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A COMPARATIVE EVALUATION OF ADVANCED CONVERTERS

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Reactor Division

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JANUARY 1965

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FOREWORD

The evaluation reported herein was begun near the end of 1963. At that time, studies were underway within the AEC to determine which of the various heavy-water-moderated reactor concepts should be pursued. This comparative evaluation of advanced converters was in its final stages of completion when the commission decided to focus the development effort for heavy-water-moderated reactors on an organic-cooled system. The data presented in this report are for the pressurized heavy-water cooled case, whereas the heavy-water reactor studies referred to above indicated that the organic-cooled system has better economic potential. Thus the heavy-water reactor concept evaluated in this report differs from the particular reactor proposed for development by the AEC. The significant advantages of the organic coolant compared with heavy-water coolant are said to include higher coolant temperatures, with resulting higher thermal efficiency, easier adaptability for on-line refueling (because of the much lower coolant pressure), and a significantly lower heavy-water inventory.

This report was planned originally to include an evaluation of a 1000-Mw(e) seed-blanket reactor. However, the only detailed design available was for a 500-Mw(e) plant, which was designed for long life between refuelings. The ground rules for this study include the use of 10% annual charges on fuel inventory as the reference case, while the particular seed-blanket design available had been developed for low-interest-rate financing conditions. As the study progressed, it became evident that the long-lived core design being evaluated was especially sensitive to the assumption of 10% financing charges on the fuel inventory. Because of this and the fact that a design optimized for the ground rules of this study was not available, it was decided that inclusion of the seed-blanket reactor evaluation in this final report was inappropriate. Since completion of this report the expected performance of a 1000-Mw(e) seed-blanket breeder reactor concept better optimized for the ground rules of this study has been estimated, but time has not permitted including this recent information.

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1. INTRODUCTION

The majority of the reactor concepts under development in the United States for central station power generation may be considered to fall into one of three categories: pressurized-water reactors (including boiling-water reactors); advanced-converter reactors; and breeder reactors. The usefulness of this particular classification of power reactors relates to the time periods in which reactors of each category are likely to be built. Most forecasts of the pattern in which the U.S. nuclear economy will develop postulate that an initial period of pressurized- and boiling-water reactor construction will ultimately be followed by an era of breeder reactor dominance. Disagreement exists, however, on whether there will be an intermediate period in which the advanced converters will hold sway and, if so, which of the various converter types is most attractive for that period.

The advocates of advanced converters point out attributes possessed by particular members of the class: good uranium utilization, use of thorium, low fuel cycle cost, high efficiency, long fuel life, high plutonium production (for providing fast-reactor inventories), and others. This study was initiated by the USAEC mainly to analyze the economic value of these features and to provide information on the relative merits of the various advanced converters. A design for a 1000-Mw(e) pressurized-water reactor was included in the evaluation in order to provide a comparison with advanced converters.

Bases for Comparison of Concepts

The status of development of the advanced converters is such that they will not be brought on line in any quantity before 1970; and, indeed, it is questionable whether large, ~1000-Mw(e) units can be built by 1970. On the other hand, the U.S. program of fast-reactor development is aimed at having an economical 1000-Mw(e) fast breeder in operation by 1989. Postulating that breeder construction will begin to dominate in about 25 years, the period of particular interest for advanced converters is the two decades following 1970. This study has been directed, therefore, toward determining which converters are the most attractive for construction during that period.

In order to judge "attractiveness," it is necessary first to define what attributes of a reactor are most desirable. These differ, of course, depending on the application, but for central station electricity production, a single factor that encompasses all others is plant economics. Although some subjective factors enter into their evaluation, private utilities installing new capacity will generally build the plant they believe will yield the lowest power cost. Other considerations, particularly the conservation of fuel resources, may be important in establishing national objectives, but such considerations will influence a utility only if they affect its economics. Of course, in evaluating economics, future conditions must be properly weighted and attitudes of governmental agencies,

as reflected in the regulation of financing and rates and in tax provisions, taken into consideration.

Because economics will be the primary factor affecting choice of reactor type for construction by a utility, the major effort of this study has been directed toward estimating what the cost of power from each reactor concept would be if a number of reactors of that design were built in the period following 1970. As described later, the procedure has been to secure a design for a 1000-Mw(e) reactor from the sponsor or developer of each concept and to estimate its capital, operating, and fuel-cycle cost. In order to ensure that the cost comparisons of the different systems are valid, it has also been necessary to establish that the postulated performance is achievable and to identify the developments required. To do this the design and predicted engineering performance of the reactor plants were subjected to review and evaluation.

Reactor Systems Studied

The concepts evaluated were pressurized-water, spectral-shift-control, pressure-tube heavy-water, high-temperature gas-cooled and sodium-graphite reactors. Design details, obtained from the sponsors of each concept, are summarized in Chapter 3. The concepts, the sponsors, and the type of information obtained from each are indicated briefly in the following paragraphs.

Pressurized-Water Reactor (PWR)

The PWR considered was a 1000-Mw(e) uranium-fueled reactor, with soluble-poison control, which was based on a design study by Westinghouse Atomic Power Division. Because the Westinghouse study was completed before this evaluation began, no changes were made by them to fit the ground rules of this study. It was mutually agreed, however, for the purposes of this evaluation to substitute Zircaloy fuel cladding for the collapsed stainless steel cladding that was used in their reference design.

Spectral-Shift-Control Reactor (SSCR)

The SSCR is a 1000-Mw(e), thorium-fueled reactor cooled and moderated by an H_2O-D_2O mixture whose concentration is changed progressively during core life. The evaluation study was based on a core design submitted by the Babcock & Wilcox Corp. Limited plant design and capital cost information was also provided by B & W, and they were consulted about specific points which arose.

Pressure-Tube Heavy-Water Reactor—Uranium (HWR-U); Pressure-Tube Heavy-Water Reactor—Thorium (HWR-Th)

Heavy-water-moderated and pressurized-heavy-water-cooled reactors of 1000-Mw(e) capacity were considered. The evaluation was based on core

designs provided by E. I. Du Pont de Nemours & Company (Savannah River Laboratory). Capital cost data and some plant design information were also obtained from Du Pont.

High-Temperature Gas-Cooled Reactor (HTGR)

The HTGR is a 1000-Mw(e), helium-cooled, thorium-cycle reactor using fuel-moderator elements of graphite with BeO spines. The evaluation was based mainly on the "TARGET" concept of General Atomic. While plant design and capital cost information were derived from General Atomic reports, additional information on core conditions and selection of fuel management was provided directly by GA.

Sodium-Graphite Reactor (SGR)

The SGR is a 1000-Mw(e) "calandria" reactor with vented fuel elements of Zircaloy-clad uranium carbide. The evaluation was based on design and capital cost information in Atomics International reports, with, however, substitution of fuel element design and fuel management information that was provided directly by AI.

Procedures Followed

At the initiation of the study, the time period and economic context in which the reactors were to be evaluated were defined and a set of ground rules was formulated that took into consideration suggestions and comments by the AEC and the sponsors of the various reactors. The major provisions of the ground rules, which are given in detail in Chapter 2, are:

1. The technology to be used in the reactor plants was generally restricted to that which would be available for a smaller, prototype reactor to be built for startup in 1970.
2. Economic parameters were selected to represent the average conditions existing during the life of reactors built in the period 1970-1980. Private ownership of reactors, fuel, and fabrication and reprocessing plants was assumed.
3. The total electrical capacity of the reactors was specified as being the same for each type; consequently, the fuel fabrication and processing industry size varied with the fuel burnup.
4. The fuel-cycle cost was estimated by using present value accounting and following the behavior of the system over a 30-year period.
5. It was considered that the uranium from thorium systems would be recycled throughout the reactor lifetime, but the sale of plutonium with no recycle was assumed.

The reactor sponsors were asked to submit core designs based on these ground rules for review and to provide plant designs and operating and capital cost data. This information was received throughout the first half of 1964.

In order to make the comparisons as meaningful as possible, the study was organized by function, with one group performing all the physics analyses, another making all the processing cost estimates, etc. Coordinators for each reactor concept were appointed to serve as contacts with the reactor sponsors, to specify the conditions for the concept, and to disseminate information to the functional groups. The organization of the study is shown in Table 1.1.

Table 1.1. Organization of the Study

Study Coordination

M. W. Rosenthal

Reactor Physics

L. L. Bennett, D. R. Vondy, C. W. Craven, E. H. Gift (cross-sections), L. Jung, W. E. Thomas, Rachael Hackney (Nuclear Utility Services), R. Sullivan (Nuclear Utility Services)

Engineering Evaluation of Reactor Cores

R. C. Olson; R. E. Adams, J. L. Scott, J. R. Weir (fuel element performance); C. G. Lawson, J. C. Moyers, J. L. Wantland (thermal design)

Plant Evaluation and Reactor Capital Costs

R. C. Olson, C. W. Collins, R. E. Hoskins, M. L. Myers

Fuel Fabrication Costs

D. A. Douglas, A. L. Lotts, R. L. Pilloton

Fuel Processing Costs

J. T. Roberts

Fuel Shipping Costs

R. Salmon

Fuel-Cycle Costs

W. L. Carter, L. G. Alexander, L. L. Bennett, D. R. Vondy

Reactor Concepts Coordination

PWR.....C. G. Lawson
SSCR.....R. C. Olson
HWR'S.....C. G. Lawson
HTGR.....R. S. Carlsmith
SGR.....D. B. Lloyd

Upon receipt of design information, the thermal conditions, fuel element performance requirements, and other aspects of the core engineering were checked and evaluated for each reactor. Design features of the entire power plant were then reviewed. Taking into consideration differences in design requirements, as revealed by the review, adjustments were made in the cost data supplied by the sponsors, and normalized estimates of the capital costs for all the plants were produced. Normalized operating and maintenance costs were also estimated.

Physics calculations were performed for each concept with enrichments and material balances obtained for various lifetimes. Simultaneously, fuel fabrication, shipping, and processing costs were estimated as functions of plant capacity. The physics results and cost estimates were then used to obtain fuel-cycle costs, and the minimum costs were determined. Since some cost parameters are particularly uncertain (e.g., plutonium value, fuel inventory charges, and fixed charges on processing plants), values were computed and optimized for different cost assumptions. It should be noted that while the optimization procedure determines the minimum fuel-cycle cost for the core design considered, the fuel element dimensions, fuel-to-moderator ratio, plant design conditions, etc., were those specified by the reactor sponsor, and such factors were not varied.

In presenting the results in this report, separate chapters are devoted to the major subdivisions of the study. The engineering appraisals of the core and of the plant are presented in different sections, as are the capital, operating, fuel fabrication, shipping, and processing costs estimates and the physics analyses. A chapter on fuel-cycle costs combines the results of the physics and unit cost studies, and the optimum fuel-cycle costs are, in turn, added to the operating and capital costs in a chapter on power costs. Fuel utilization is discussed in a separate chapter. The major results of the study and the most important conclusions are summarized in the final chapter.

Many details that were omitted from the body of the report are provided in the Appendices. These may be useful for checking the procedures employed or as basic information for use in other evaluations.

Limitations of the Study

The objective of this study was to compare reactor concepts and not to evaluate particular designs. Nevertheless, the procedure actually followed was to compare specific designs selected for that purpose by sponsors or developers of the reactors. The justification for this approach is that the organization providing the design in each case is the most experienced in the U.S. with that type of reactor, and hence their design should be the best choice that could be made today. It is recognized that this assumption is not completely valid, since the evaluation sometimes indicated design changes that could benefit a concept. Yet, these changes usually were not major, and the advantages from them were granted in the cost evaluation where it was reasonable to do so. Hence, the procedure generally appears adequate for the comparison of costs.

This evaluation was restricted largely to comparing the power costs of reactors in the specified economic context. While some economic factors were varied in the evaluation, this was done to allow for the uncertainty in their values and not to shift the period or economic environment. Of particular importance in the cost comparison are the assumptions that (1) the low cost uranium reserves are not exhausted during the life of the reactors, and (2) the fabrication and processing plants are of a size that will support ten to fifteen 1000-Mw(e) reactors of a given type. The first assumption excludes the inherent emphasis on fuel utilization that would be a factor if reactors were evaluated in a period of high ore cost. Because of the second assumption, unit costs are well below those existing today; however, they are not so low as they will be a few decades later when the industry is even larger.

As a consequence of the ground rules, the reactor concepts were optimized largely for the production of power at the lowest cost. In all cases, better fuel utilization could be achieved at the penalty of greater power cost by altering the design and operating conditions. Hence, specific indices of fuel utilization should not be extracted from this study and used to judge the relative merits of the concepts in conserving fuel. If fuel utilization is to be used as a measure of merit, this must be specified to the reactor designer in advance. Shortening fuel exposure to observe the improvement in breeding ratio is not an adequate way of determining the conversion potential of a concept, since different fuel-to-moderator ratios or features such as fertile blankets, on-stream refueling, fertile control poisons, and others can be employed if cost is not controlling.

In estimating costs, successful development of components required by the design was postulated. The most common example is the specification of equipment that is larger than any now being built, although there are some features that have not been demonstrated even in smaller size. Development needs are identified in the engineering evaluation chapters, and the extent to which they represent extrapolations of technology is noted. Differences in the state of development of the various reactors, however, are not reflected in the plant capital cost or the fuel cycle cost estimates.

Any comparison of systems that are incompletely developed is restricted in accuracy by the state of knowledge of technical and economic matters. However, having functional groups evaluate the features of all the concepts simultaneously offered hope that the comparisons would be valid. This applies particularly to the cost estimates in this study, where there can be much more confidence in the ordering of the systems than in the absolute values. An attempt was made to recognize the weak points in the estimates, and these are identified throughout the report.

Acknowledgments

The capital cost normalization for this study was made jointly by Sargent & Lundy, Engineers, and ORNL, with Sargent & Lundy providing most of the actual cost values used. Sargent & Lundy also estimated the operating costs.

Assistance in performing the physics analyses was secured from Nuclear Utility Services, who provided two physicists to work for a period in Oak Ridge.

As noted earlier, design and cost information was supplied by the following sponsors of the reactor concepts, whose assistance and cooperation is acknowledged: Atomics International, Babcock & Wilcox Company, E. I. Du Pont de Nemours & Company (Savannah River Laboratory), General Atomic, and Westinghouse Atomic Power Division.

The authors wish to acknowledge the valuable advice and assistance received from J. A. Lane in the conduct of this study and the preparation of this report.

2. GROUND RULES AND COST BASES

Source of Data

Information on the plant designs and fuel cycles was obtained from the companies and laboratories that are the developers of the various concepts. The reports and letters received are listed in the references of Chapter 3.

General Provisions

The technology to be used in the reactor and power plant designs was, in general, restricted to that which would be available for smaller, prototype reactors to be built for startup in 1970. The reactors evaluated were presumed to begin operating in the period 1970-1980 using unirradiated fuel. The plant lifetime was chosen to be 30 years, while fuel costs were based on the integral number of fueling cycles that came closest to 30 years. It was considered that the uranium and thorium from thorium systems would be recycled throughout the reactor lifetime,* but the sale of plutonium without recycle was assumed in the uranium systems studied.

Power Cost Components

Fuel-Cycle Cost

The fuel-cycle cost was resolved into the following components: (1) burnup cost, (2) fabrication cost, (3) processing cost, (4) shipping cost, (5) fixed charges on fissile and fertile inventories, and (6) interest charges on operating capital invested in fabrication, processing, inventories of special nuclear materials, and shipping. The fixed charges were computed by the "present worth" formula. A value was assigned to fuel discharged from the reactor at the end of plant life (~30 years), including linearly prorated credits for fabrication and debits for processing of partially irradiated fuel. In computing the present worth of the power produced, it was assumed that the price of power and the plant factor would not change with time.

Reactor Plant Capital Cost

Capital costs were estimated for each reactor plant. It was assumed that the plant was one of a number of the same type to be built and that

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*Sale of a high-burnup part of the fuel discharged from each HTGR cycle was treated, but a reduced value was used because of the large concentrations of higher uranium nuclides.

the equipment and system for the plant had been fully developed. However, discount credit for quantity orders of equipment was not assumed.

Capital cost breakdowns were arranged in accordance with the system of accounts given in the AEC Handbook "Guide to Nuclear Power Cost Evaluation," TID-7025 (Vol. 1). Indirect costs appropriate to the reactor size were estimated based on the breakdown used in TID-7025 (Vol. 1). However, new estimates were made of percentages applied for each indirect cost item to reflect recent experience. Fixed charges on the reactor plant were taken as constant at 12% per year over the plant life, as described in Appendix A.

Reactor Plant Operating Cost

Operating and maintenance costs were estimated for each reactor plant.

Financing Conventions

Private ownership of fuel and of fabrication and reprocessing plants was assumed. Because of uncertainty as to what they will actually turn out to be, various values of fixed charges and interest rates were used, as indicated in the following paragraphs. For the same reason, the sale price of plutonium was treated as a parameter. Single values of each quantity were specified, however, for use in selecting reference designs.

Ownership of fissile and fertile materials during fabrication and processing, as well as when on site at the reactor, was considered to be vested in the reactor plant. Inventory charges on fissile and fertile inventories were computed in the range from 6 to 12% per year, with a reference value of 10% per year. For determining the inventory charges, changes from initial to final values were assumed to occur linearly with time during irradiation.

Interest charges on the fabrication cost of fuel elements were computed in the same way as the fixed charges on fuel. For this purpose, the fuel elements were assumed to depreciate linearly with time over the period of irradiation. The discount factor for computing present worth was 6% per annum. Interest payments and collection of revenue were assumed to be made twice a year.

Processed fuel was assumed to be utilized immediately in a "ready" reactor of the type being studied. This provision relieved a reactor of inventory charges on idle fuel for the period between the end of processing and the start of fabrication. Recycle fuel of the same composition and amount as that discharged (after subtracting processing losses) was assumed to be available for use with makeup U^{235} in the next cycle of the batch-loaded SSCR. However, for the initial refuelings, unirradiated uranium was fed to the HWR-Th and the HTGR throughout the time required to process and refabricate the discharged fuel.

Value of Materials

1. Natural uranium as U_3O_8: \$8.00/lb U_3O_8.
2. Unirradiated thorium as ThO_2: \$5.00/lb ThO_2.

3. The value of unirradiated enriched uranium was based on a separative work cost of $30/kg.

4. Depleted uranium of low enrichment: value corresponding to its enrichment based on total uranium present, with no additional penalty for U^{236} content.

5. Irradiated enriched uranium: $12/g of fissile isotopes as nitrate (except that only $7/g was allowed for the highly irradiated fuel sold after each HTGR cycle).

6. Plutonium: variable parameter, in the range $3 to $10 per gram of fissile isotopes as nitrate, with a reference value of $7.00.

7. D_2O: $20/lb.

8. Conversion of U_3O_8 to UF_6: $2.70 per kilogram of uranium.

Fabrication Plant

The fabrication plant was considered to be centrally located at the same site as the processing plant and to utilize recycled thorium and uranium (including makeup) as the oxides from the reprocessing plant. Fixed charges on depreciating capital were treated as a parameter, with values from 15 to 30% per annum and a reference value of 22% per annum.

Oxide fuels were assumed to be fabricated by vibratory compaction of particles prepared by the sol-gel process in the reprocessing plant. Coated carbide particles were assumed to be manufactured from oxide particles prepared by the sol-gel process in the reprocessing plant.

Two different plant arrangements were studied: (1) a single-purpose fabrication plant with capacity for serving an electrical industry of 15,000-Mw(e) capacity and (2) a dual-purpose fabrication plant capable of serving 10,000 Mw(e) of PWR's and 10,000 Mw(e) of the concept being studied.

The price of fabricated fuel was assumed to remain the same throughout the 30-year plant life. The design of the fuel elements was specified in detail by the proponents. Holdup time was established for each concept. Losses of thorium and uranium were assumed to be 0.2% per pass.

Reactor Plant

The electric station was required to have a net capability of 1000 Mw(e), with one reactor per station. The condenser pressure was assumed to be 1 1/2 in. Hg abs. The reactor plant availability was required to be 0.9 in order to achieve an average plant factor of 0.8 over the plant life.

Fuel was considered to be received at the reactor site 60 days before loading. The cooling time before shipment of irradiated fuel was established for each concept.

Reprocessing Plant

The reprocessing plant was assumed to be centrally located at the same site as the fabrication plant. The UF_6 (purchased for makeup) was assumed to be reduced in the reprocessing plant; thorium and uranium were returned to the fabrication plant in the form of oxide particles prepared by the sol-gel process.

Losses were assumed to be 1% per pass for thorium, uranium, and plutonium. Holdup time was established for each concept.

Fixed charges on depreciating capital were treated as a parameter, with values from 15 to 30% per annum and a reference value of 22% per annum.

Two different plant arrangements were considered: (1) a single purpose reprocessing plant having the capacity for serving 15,000 Mw(e) of a given concept and (2) a dual purpose reprocessing plant serving 10,000 Mw(e) of PWR's and 10,000 Mw(e) of a given concept, with a single solvent-extraction line but separate head-end plants, where required.

The price for reprocessing was assumed to remain the same during the life of the reactor. Protactinium losses were based on a 180-day decay time from the time of discharge from the reactor.

Shipping

It was assumed that fabrication and reprocessing would be performed at the same site, which was located 1000 miles from the reactor. Casks were assumed to cost $1.00 per lb of cask weight. The maximum cask weight was taken as 110 tons fully loaded. The cask utilization factor was estimated for each concept. The cask handling fee used was $500 per round trip, and the cask life was taken as 30 years.

The freight rates assumed were the following: loaded cask, $0.0193 per lb; empty cask, $0.0181 per lb. Insurance against property loss was charged at 0.4% of the value of each shipment. The cost of liability insurance was included in the charges against the reactor, processing, and fabrication plants.

3. DESCRIPTION OF REACTORS

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Brief descriptions of each of the six reactor designs are given in this chapter. Wherever possible the descriptions are based on reports of the organization that is the sponsor of the reactor type. In most cases the reports were supplemented by information obtained from correspondence and informal discussions. During the course of the evaluation, it was necessary to change some of the reactor design parameters in order to place each of the designs on a comparable basis, and there was some reoptimization of the fuel cycle. Hence there are numbers in this chapter that are not in exact agreement with those given in the referenced reports.

Pressurized-Water Reactor (PWR)

The pressurized-water reactor[1] is a 1000-Mw(e) reactor fueled with slightly enriched UO_2 and both moderated and cooled with pressurized H_2O. The design characteristics of the reactor are presented in Table 3.1.

The reactor is cooled by seven loops, with one pump and one steam generator per loop. Flow through the core is single-pass upward. The active core is 11 ft high and 12.8 ft in diameter. The reactor contains 264 fuel assemblies and 109 control rods. A cross section drawing of a fuel assembly and cruciform-shaped control rod is shown in Fig. 3.1. (For the purposes of this study the fuel cladding was changed from 0.010-in.-thick stainless steel to 0.020-in.-thick Zircaloy-4 with the cladding outside diameter remaining constant.) The fuel assembly contains a 16 × 16 square array of rods. The rods with the highest specific power near the control rods are of a reduced diameter. The cruciform control rods have short Zircaloy followers that promote coolant flow toward the fuel pins with the highest surface heat flux. Chemical poison control in the form of boric acid solution is used for shim control to reduce the control rod requirements.

Fuel management consists of three-zone scatter reloading, which results in an in-core lifetime in the reference design of 2.3 years for the fuel. One third of the core fuel is replaced each 9 months at a 0.8 plant factor.

The peak-to-average heat flux in the PWR is 3.33 and the core power density is 80.7 kw/liter. No special methods are used to reduce the peak-to-average power density.

Table 3.1. Summary of Reactor Characteristics

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Reactor power						
Core nuclear power, Mw(th)	3220	3266	3747	3870	2270	2336
Net electrical power, Mw	1002	1000	1003	1010	1008	1019
Net thermodynamic efficiency, %	31.1	31.2	26.8	26.1	44.4	43.6
Coolant						
Composition	H_2O	Variable: 80% D_2O and 20% H_2O to 2% D_2O and 98% H_2O	D_2O (99.75%)	D_2O (99.75%)	Helium	Sodium
Core inlet pressure, psia	2050	2200	1900	1874	450	48
Core pressure drop, psi	34	22	43	24	3.9	21
Flow rate, lb/hr	160×10^6	146×10^6	138×10^6	138×10^6	8.32×10^6	66×10^6
Core inlet temperature, °F	546	535	507	507	720	750
Mean core outlet temperature, °F	598	594	579	579	1470	1150
Number of coolant loops	7	5	6	6	6	5
Core thermal performance						
Average heat flux at rated power, Btu/hr·ft^2	163,000	253,000	458,000	326,000	74,000	338,000
Maximum heat flux at rated power, Btu/hr·ft^2	544,000	470,000	820,000	621,000	140,000	760,000
Maximum fuel pin surface temperature, °F		650	626	626		
Peak-to-average power ratio	3.33	1.9	1.55	1.55	1.35	2.25
Average core power density, kw/liter	80.7	130	21.4	16.05	7.0	11.91
Average specific power, kw/kg of fertile material	31.7	49.3	55.7	48	27	32
Description of core and fuel						
Moderator material	H_2O	D_2O and H_2O	D_2O (99.75%)	D_2O (99.75%)	Graphite and BeO	Graphite
Reflector material	H_2O	D_2O and H_2O	D_2O (99.75%)	D_2O (99.75%)	Graphite	Graphite
Fuel material	UO_2	UO_2 and ThO_2	UO_2	ThO_2 and UO_2	UC_2 and ThC_2	UC (hyperstoichiometric)
Equivalent diameter of core, ft	12.8	10.17	22.4	25.83	31	25.1
Active height of core, ft	11.0	10.92	15.0	15.0	15.5	14
Number of fuel assemblies	264	156	516	688 (88 blanket rods in reflector)	5489	594
Fuel element spacing, in.	8.4	8.656	10.0	10.0	4.70	10.25
Fuel rod outside diameter, in.	0.378 and 0.320	0.420				0.590
Cladding material	Zircaloy-4	Zircaloy-4	Zircaloy-2	Zircaloy-2	None	Zircaloy-2
Cladding thickness, in.	0.020	0.0255	0.025	0.025		0.020

Table 3.1 (continued)

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Description of fuel cycle						
Fuel management scheme	Three-batch scatter reloading	Zoned enrichments, single-batch reloading	Three-batch scatter reloading (power-flattened zone)	On-line refueling (power-flattened zone)	12-batch scatter reloading, axial zone loading	1/6 of core reloaded, 50 elements relocated
In-core residence time at 0.8 load factor,[a] yr	2.3	2.1	1	1.6	6	3.8
Core average fuel exposure,[a] Mwd/MT	25,000	30,000	15,400	22,200	57,000	35,000
Average exposure in maximum rod,[a] Mwd/MT	30,000	45,000	19,100	27,600		40,000
Control system						
Method of shim control	Soluble poison	Variable D_2O concentration	Cruciform poison elements	Cruciform poison elements	Poison rods	Poison rods
Method of safety control	Poison rods	Poison rods	Poison rods	Poison rods	Poison rods	Poison rods
Number of rods	109	37	61	69	242	102
Rod material	Ag, In, Cd	Boron-stainless steel	Ag, In, Cd	Ag, In, Cd	30 wt % B in C	Dy_2O_3
Reactor vessel						
Type	Cylindrical	Cylindrical	Calandria	Calandria	Cylindrical	Calandria
Material	Stainless steel-clad carbon steel	Stainless steel-clad carbon steel	Zr–2 1/2% Nb tubes, type 304 stainless steel shell	Zr–2 1/2% Nb tubes, type 304 stainless steel sheel	Prestressed concrete	Type 304 stainless steel
Inside diameter, ft	15.8	13.5	25	29.17	56.5	34
Turbine plant						
Throttle steam pressure, psig	635	585	480	480	3500	3500
Throttle steam temperature, °F	495	490	466	466	1050	1000
Type of steam cycle	Saturated with reheat	Saturated with reheat	Saturated without reheat	Saturated without reheat	Supercritical with reheat	Supercritical with reheat
Feedwater return temperature, °F	443	435	387	387	538	512
Arrangement of turbine generators	Tandem compound, six flow	Tandem compound, six flow	Tandem compound, six flow	Tandem compound, six flow	Cross compound, four flow	Cross compound, four flow

[a]Reference values from concept sponsors; enrichments and fuel lifetimes obtained in this study for minimum fuel-cycle cost are reported in Chapters 11 and 12.

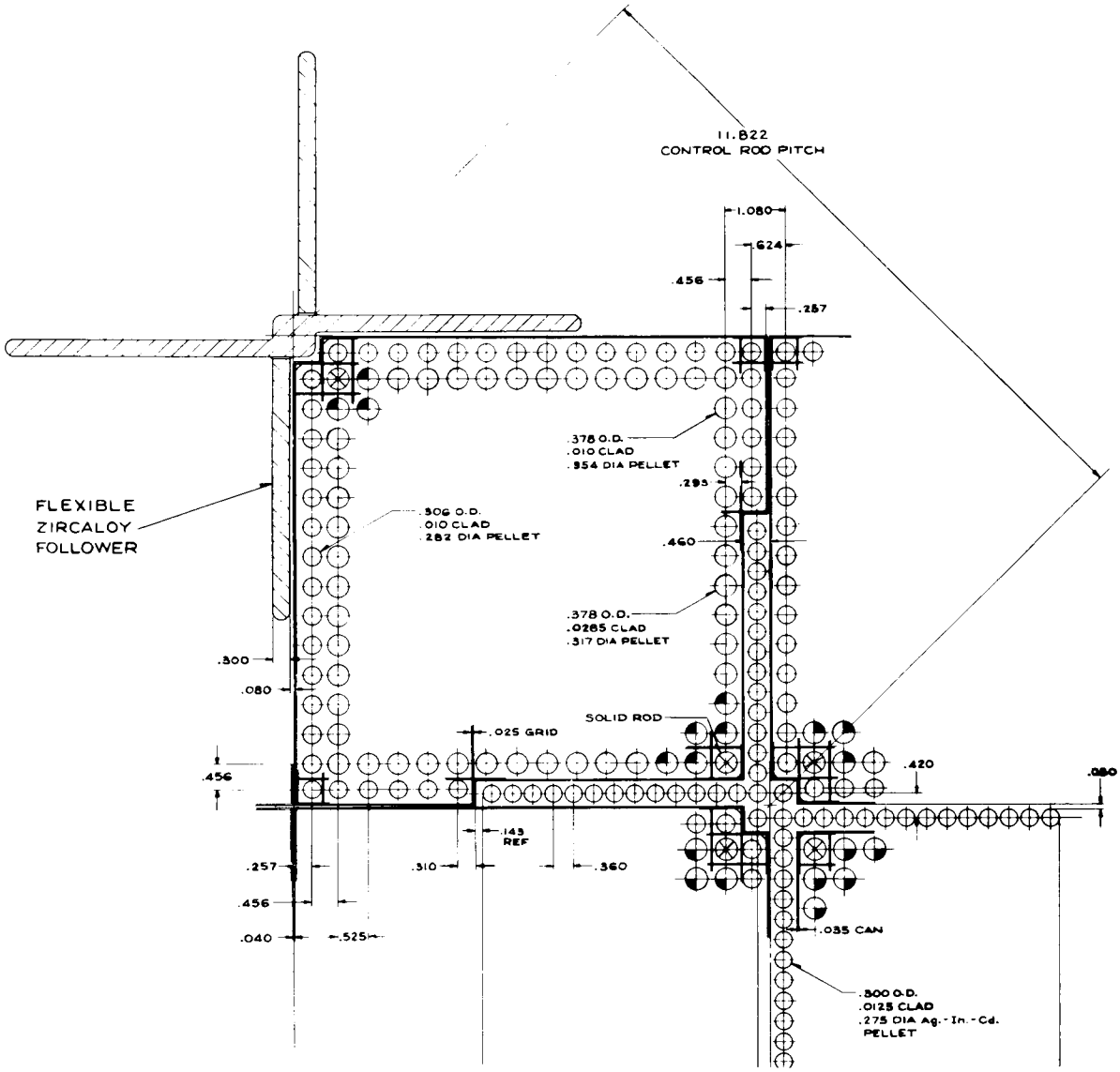

Fig. 3.1. Cross-Sectional Diagram of PWR Fuel Assembly and Cruciform-Shaped Control Rod. Dimensions in inches.

Spectral-Shift-Control Reactor (SSCR)

The spectral-shift-control reactor is a pressurized-water reactor and is similar in many respects to the poison-controlled 1000-Mw(e) closed-cycle water reactor. The principal differences in characteristics between the SSCR and the PWR are associated with the use of a mixture of heavy and light water, instead of light water, as a coolant and moderator for the core. In the SSCR, increasing the concentration of H_2O over the life of the core provides shim control by reducing the average neutron energy and thereby reducing the resonance absorptions in fertile material.

The following advantages are reported for the use of heavy water for shim control:

1. Neutrons that would otherwise be lost to poisons at the beginning of operation of a new core produce new fissile material by resonance absorption in the fertile material. This results in an improvement in the conversion ratio.

2. Distortions in power distribution are reduced by eliminating the need for shim control rods. This factor, together with proper distribution of fuel throughout the core, makes possible a high degree of power flattening over the entire core life and permits operation at a high power density.

The characteristics of the 1000-Mw(e) SSCR reference design are given in Table 3.1. Most of the information on the design parameters for this reactor was received in transmittals from the Babcock & Wilcox Company or in conversations with their representatives. There is no formal report as such on the 1000-Mw(e) design; however, the features of SSCR reactors of this size are discussed in a paper by Edlund[2] and the design of a 528-Mw(e) uranium-fueled SSCR plant was obtained[3] from B & W. Additional information was obtained in meetings and by correspondence.[4-9]

A comparison of some of the characteristics of the SSCR reference design with those of the PWR reference design serves to illustrate the main similarities and differences. The SSCR fuel element is similar to the PWR element. It consists of a rod-bundle cluster of Zircaloy-4-clad vibration-compacted UO_2–ThO_2 fuel. There are 238 0.420-in.-OD pins in an assembly and 156 assemblies in the reactor. The SSCR and PWR reactor core arrangements are similar, but the dimensions of the SSCR core are smaller. The higher power density of the SSCR, 130 kw/liter versus 80.7 kw/liter for the PWR, results in a significant decrease in pressure vessel size. Because of differences in shim control, the SSCR requires only 37 control rods, compared with 109 required in the PWR.

The SSCR uses a zoned-enrichment, single-batch-reloading, fuel-management scheme, with a 2.1-year cycle time. Both radial and axial variations in fissile material enrichment, as shown in Tables 3.2 and 3.3 and Figs. 3.2 and 3.3, are employed to achieve the required power flattening to give

Table 3.2. Fuel Pin Enrichments of SSCR Normalized to an Average Core Enrichment of 1.0

Radial Zone	Relative Enrichment[a]		
	E_1	E_2	E_3
1	0.83	0.71	0.61
2	0.88	0.76	0.61
3	1.03	0.86	0.76
4	1.39	1.20	1.03

[a]See Fig. 3.2 for location of pins.

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FUEL-PIN TYPE	RELATIVE ENRICHMENT	NUMBER OF PINS PER ELEMENT
1 (NORMAL)	E_1	205
1 (DEAD ZONE)	E_1 ON ENDS ZERO IN MIDDLE	8
2	E_2	20
3	E_3	5

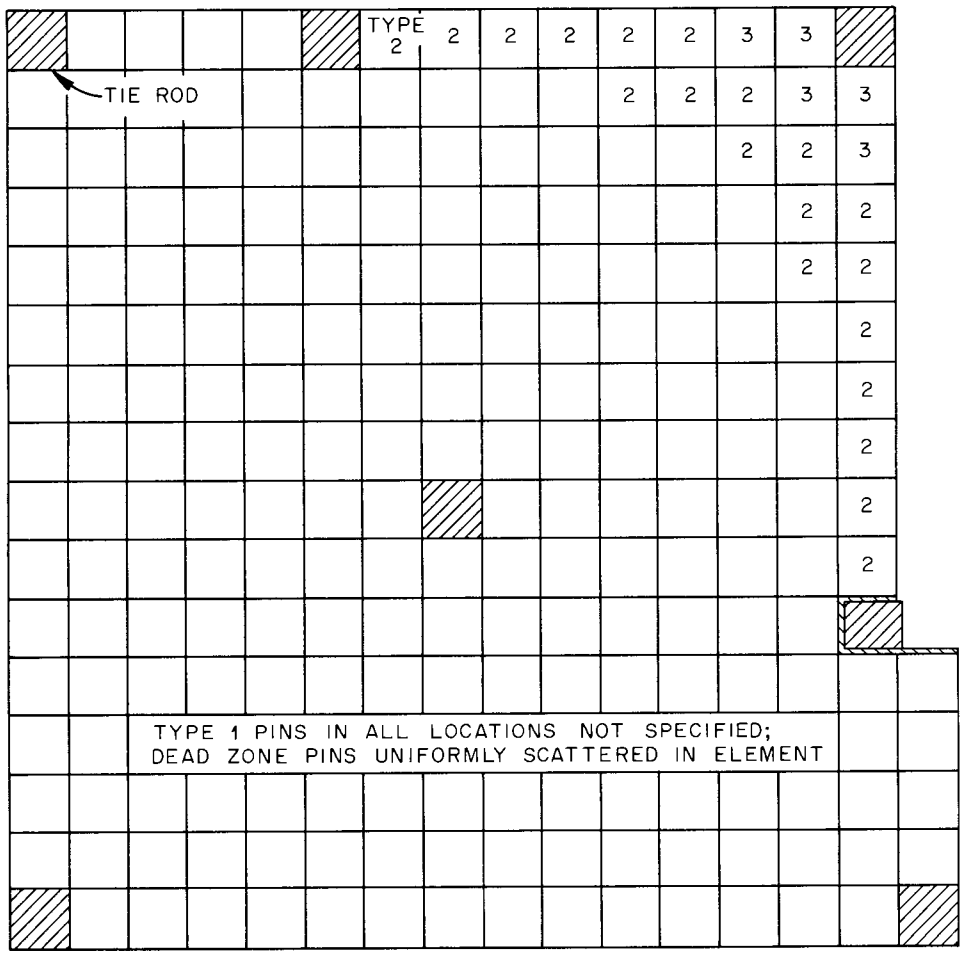

Fig. 3.2. Cross-Section Diagram of SSCR Fuel Element Showing Locations of Low-Enrichment Pins. See Table 3.2 for relative enrichments.

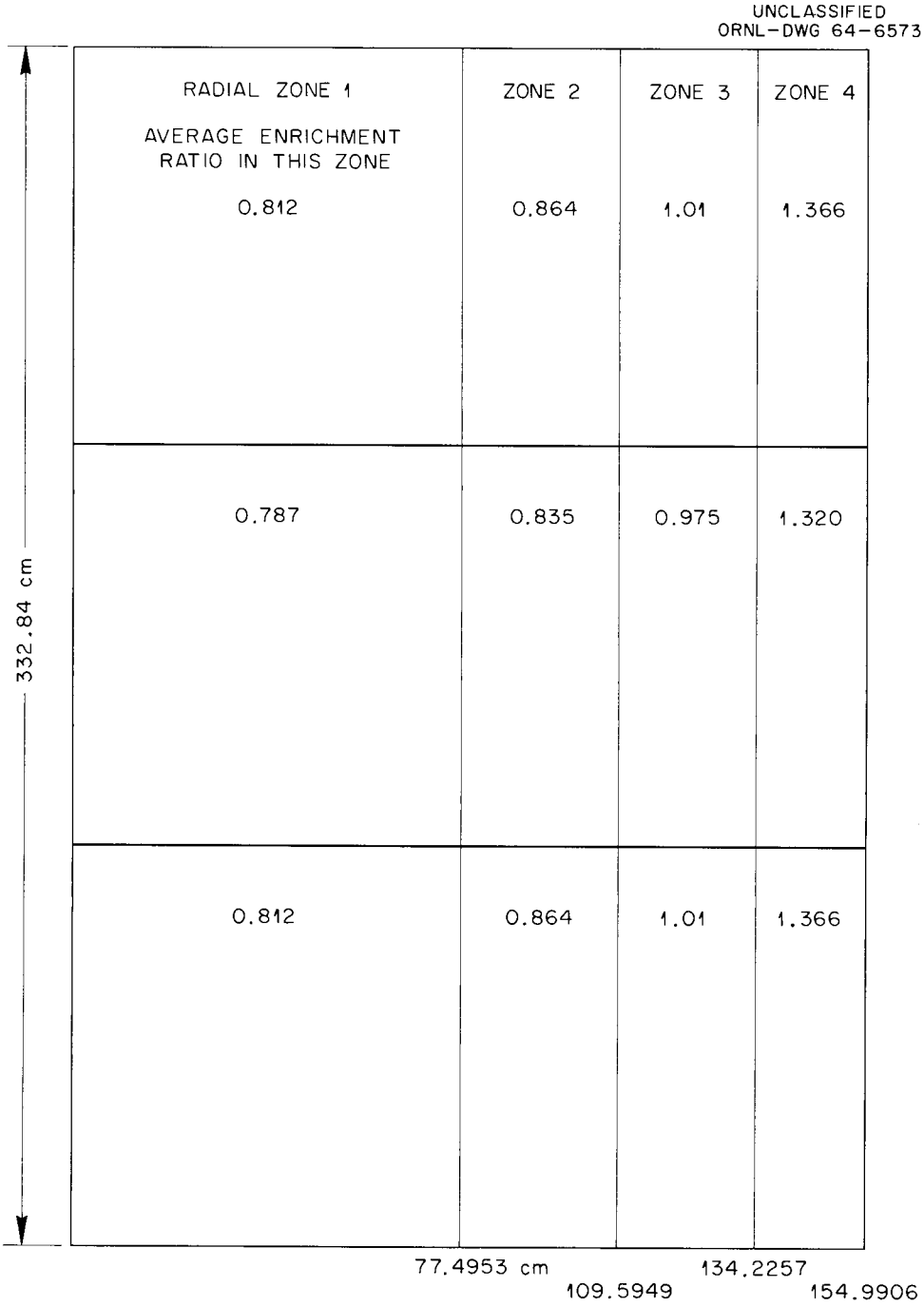

Fig. 3.3. Cross-Section Diagram of SSCR Fuel Element Showing Axial Enrichment Zones. All zones are equal volume; zone enrichment ratios are relative to core average enrichment of 1.0.

Table 3.3. Number of Fuel Pins of Various Enrichments in SSCR

Elements per zone: 39
Active pins per element: 238

	Number of Pins	
	Per Element	Per Zone
Pins with E, enrichment		
Fueled full length	205	7995
Dead zone in mid-third	8	312
Pins with E_2 enrichment	20	780
Pins with E_3 enrichment	5	195
Total	238	9282

a peak-to-average power ratio of 1.9 as compared with the PWR value of 3.33. An elevation view of the pressure vessel and core is shown in Fig. 3.4.

Pressure-Tube Heavy-Water Reactor—Uranium (HWR-U)

The pressure-tube heavy-water reactor[10-14] differs from the pressurized-water reactor in that the moderator and coolant are separated, D_2O rather than H_2O is used for both services, and the moderator is maintained cool and unpressurized. The fuel and high-pressure coolant are contained in Zr—2.5% Nb alloy pressure tubes that pass vertically through the low-pressure, low-temperature moderator tank. The primary coolant system outside the reactor is similar to that of a pressurized-water reactor except that extra precautions must be exercised to avoid heavy-water losses. This combination of D_2O and low thermal-neutron cross-section structural materials in the core permits excellent neutron economy. Design characteristics for the pressure-tube heavy-water reactor are given in Table 3.1.

The reactor vessel is about 25 ft in diameter and about 19 ft high, including axial and radial reflectors of 24 and 20 in., respectively, as shown in Fig. 3.5. It contains the D_2O moderator, which is at a nominal temperature of 60°C and substantially at atmospheric pressure, and is pierced by 516 vertical pressure-containing tubes that house the fuel and pressurized D_2O coolant. An additional 61 tubes that are not pressurized contain the control rods. The D_2O coolant enters the bottom of the calandria through individual lines from a header that circles the bottom of the reactor slightly outboard of and below the vessel. This

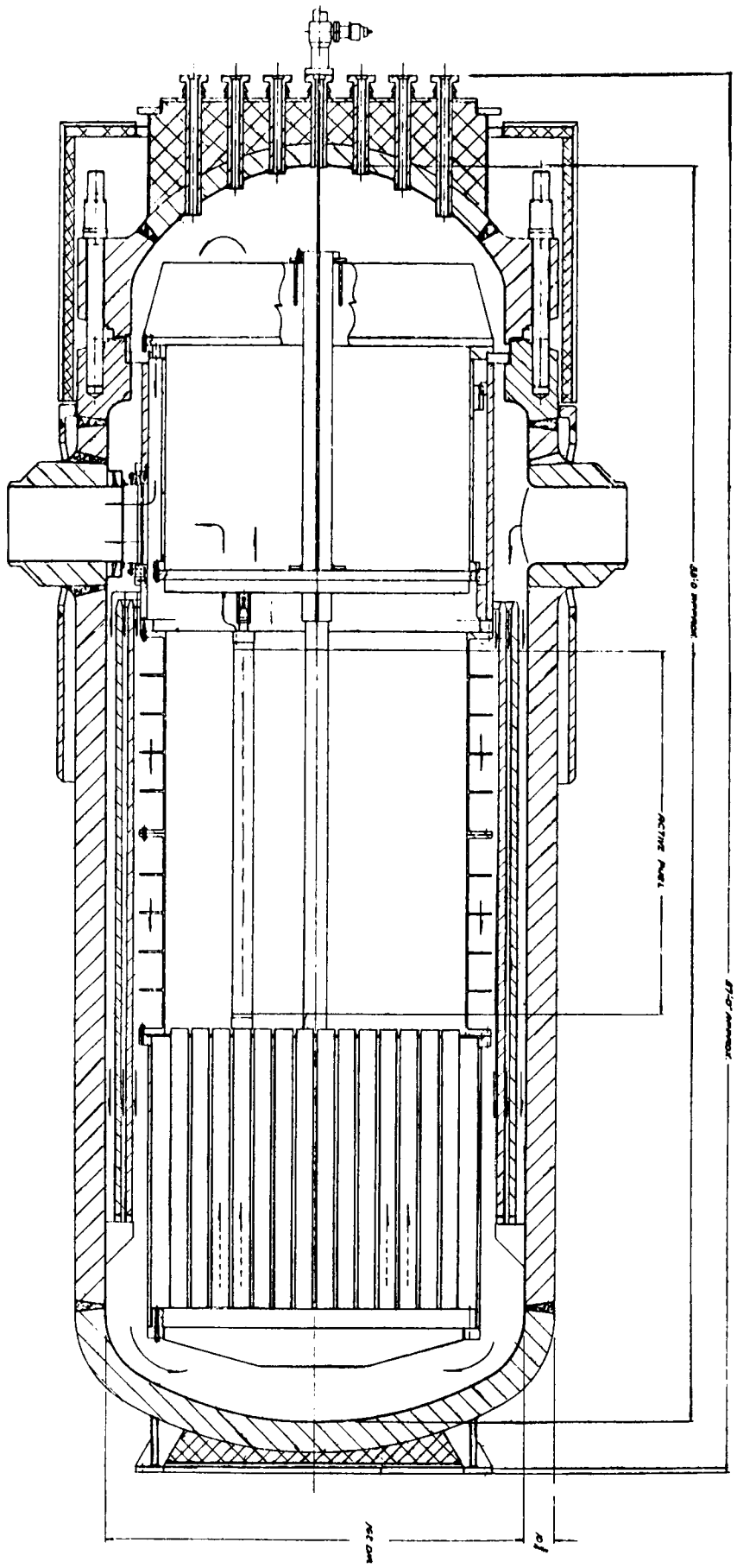

Fig. 3.4. Cross-Sectional Elevation View of SSCR Pressure Vessel and Core.

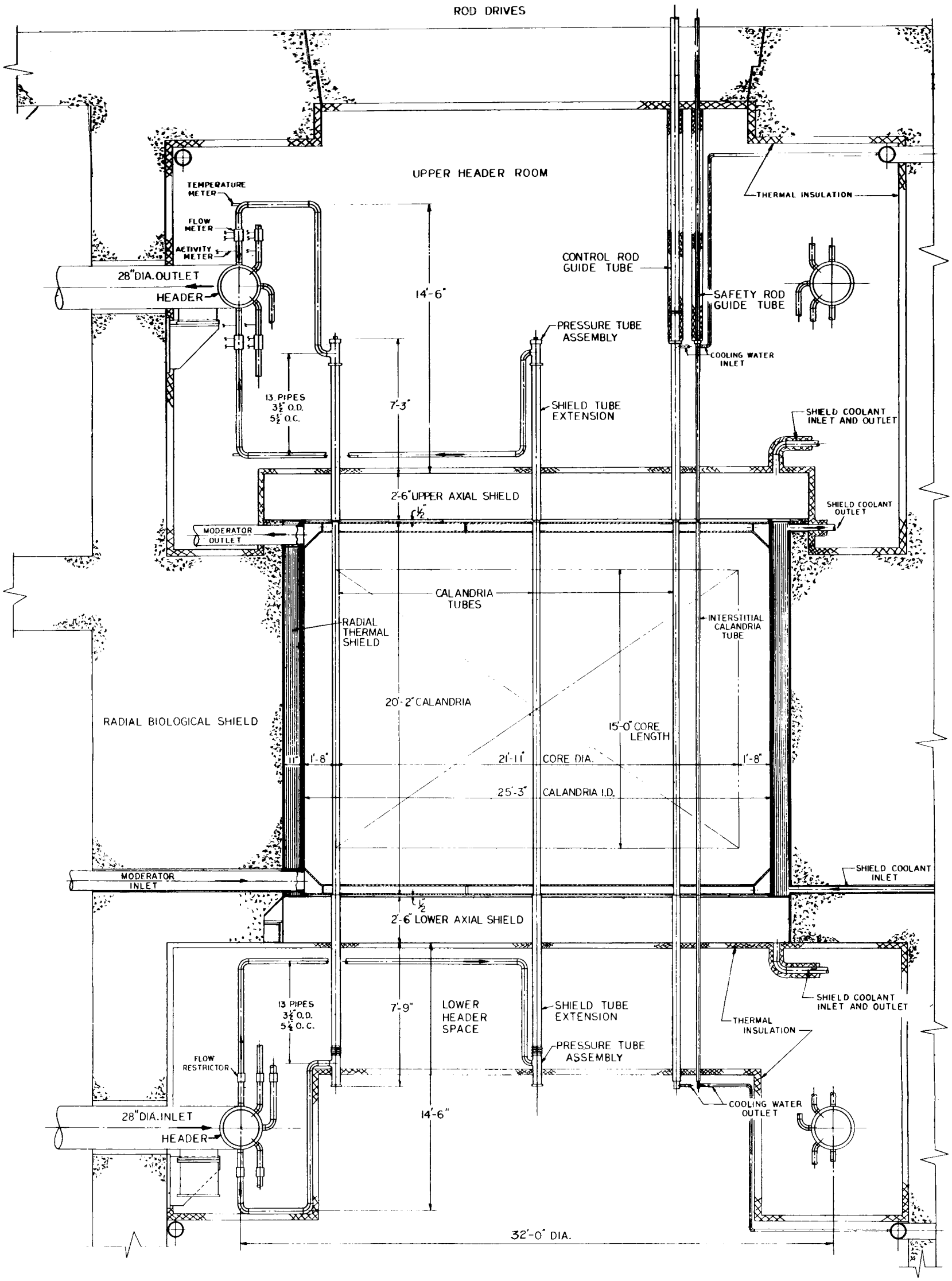

Fig. 3.5. Elevation View of Pressure-Tube Heavy-Water Reactor (HWR-U).

D_2O coolant flows upward through the pressure tubes, where it is heated by the fuel elements, and then passes through outlets located above and outboard of the calandria. The coolant is circulated to the steam generator through six parallel loops.

Several design features minimize the reactor size. The fuel assemblies are in the form of three concentrically nested annular rings, each of which has a thickness of 0.2 to 0.3 in., as shown in Fig. 3.6. These thin fuel rings permit operation at a high specific power, and a D_2O coolant velocity of about 40 ft/sec permits operation with a high surface heat flux. The 0.025-in. Zircaloy-2 cladding is not free standing. The reactor fuel management scheme includes a power-flattened zone that is maintained by control rods and a buckled zone that results in a peak-to-average specific power ratio of 1.55 and a peak assembly-to-average assembly power ratio of 1.17. The D_2O coolant channels are orificed in the buckled zone to maintain a uniform coolant temperature increase of 72°F for all fuel assemblies.

Pressure-Tube Heavy-Water Reactor—Thorium (HWR-Th)

The ThO_2-UO_2 fueled pressure-tube heavy-water reactor is similar to the uranium-fueled pressure-tube heavy-water reactor. A mixture of vibration-compacted ThO_2 and UO_2 contained in Zircaloy-2-clad concentric tubes is used as the fuel and fertile material. Single-tube thorium elements located at lattice positions in the radial reflector were specified in the original reference design but were not included in the fuel-cycle analyses of this evaluation.

While the reactor vessels for both heavy-water reactors are of the same general type, the power density of the HWR-Th is somewhat lower to reduce protactinium losses, and this results in an increase in the number of pressure tubes used to 688 and in an increase in the calandria dimensions. The use of Zircaloy-2 insulating liners reduces the pressure-tube temperature and hence the tube thickness; improved neutron economy is thereby effected. Plant thermal and electrical ratings differ slightly, and eight more control rods are required for the HWR-Th. On-stream refueling is thought to be justified for the HWR-Th to reduce neutron losses due to control rod action following shutdown refueling. Control rod insertion would be necessary to control excess reactivity buildup due to Pa^{233} decay into U^{233} during the shutdown period. In other respects the reactor is quite similar to the HWR-U (Fig. 3.5). Design characteristics are listed in Table 3.1, and Figs. 3.7 and 3.8 show some details of the lattice and fuel element, respectively.

High-Temperature Gas-Cooled Reactor (HTGR)

The high-temperature gas-cooled reactor design reviewed in this report was developed by the General Atomic Division of General Dynamics Corporation under contract with the AEC. Helium, at a pressure of 450 psi, is used as the reactor coolant. The reactor is graphite and beryllium oxide moderated and utilizes thorium in the fuel cycle. There is

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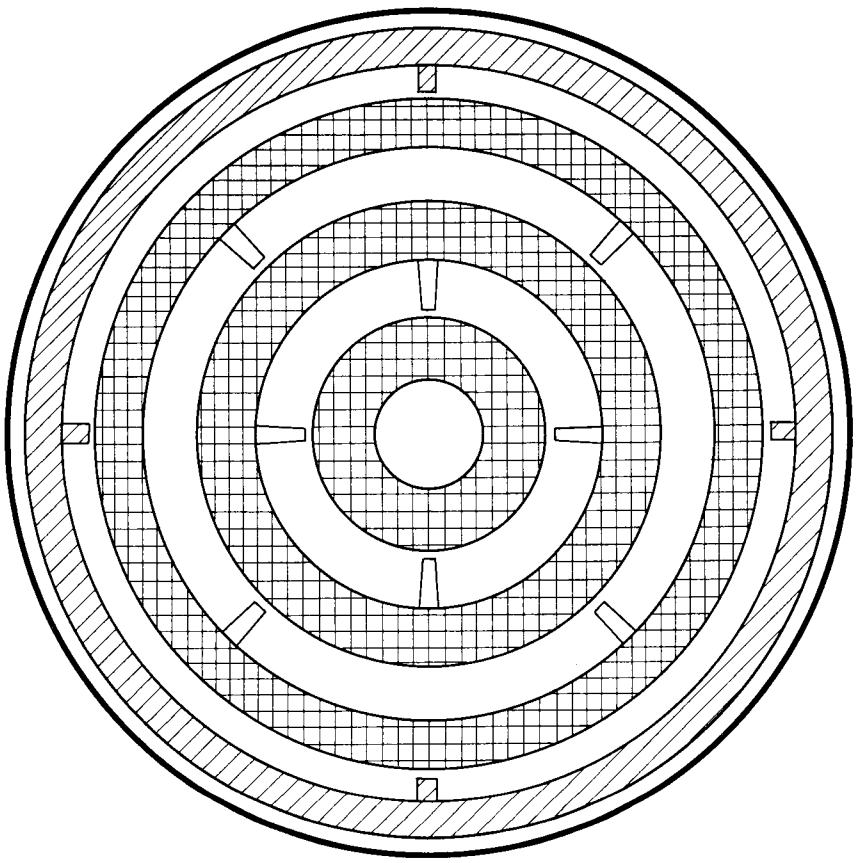
	DIMENSIONS (in.)	
	OD	ID
PRESSURE TUBE	4.468	4.043
FUEL TUBE NO. 1	3.683	3.161
FUEL TUBE NO. 2	2.564	1.922
FUEL TUBE NO. 3	1.282	0.610



ZIRCALOY CALANDRIA TUBE

Zr-2.5 % Nb PRESSURE TUBE

UO_2 FUEL TUBES CLAD WITH 0.025-in. ZIRCALOY

Fig. 3.6. Cross Section of Fuel Assembly of Pressure-Tube Heavy-Water Reactor (HWR-U).

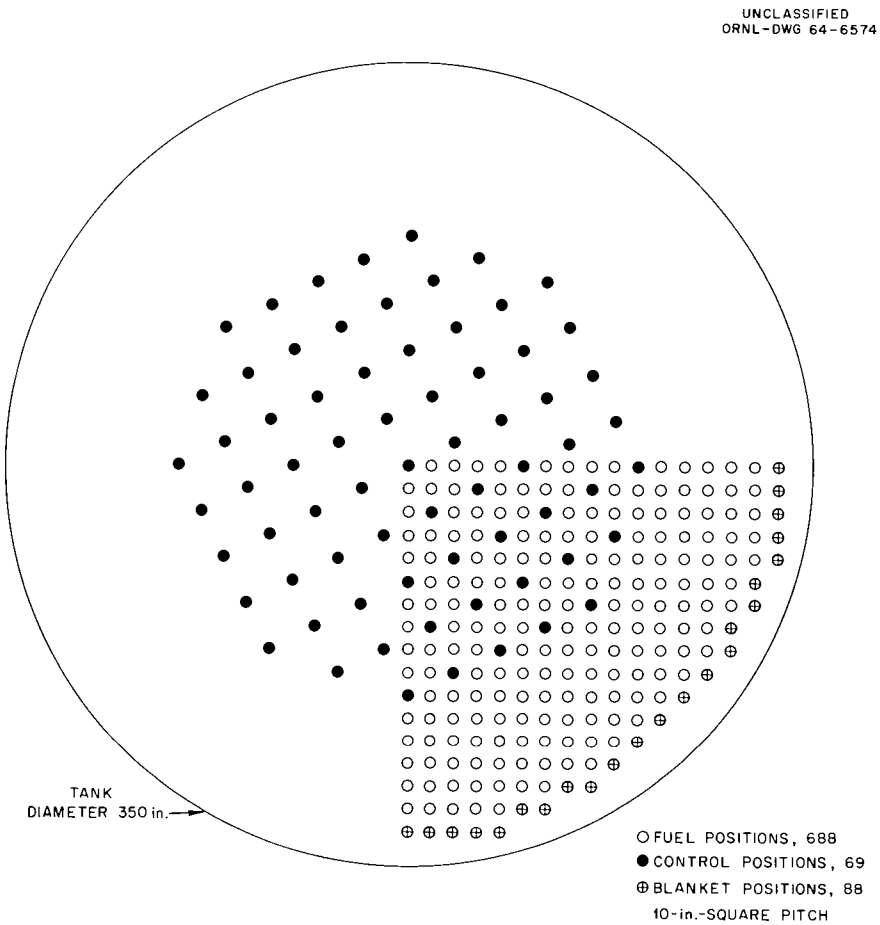


Fig. 3.7. Lattice for 1000-Mw(e) Pressure-Tube Heavy-Water Reactor–Thorium (HWR-Th).

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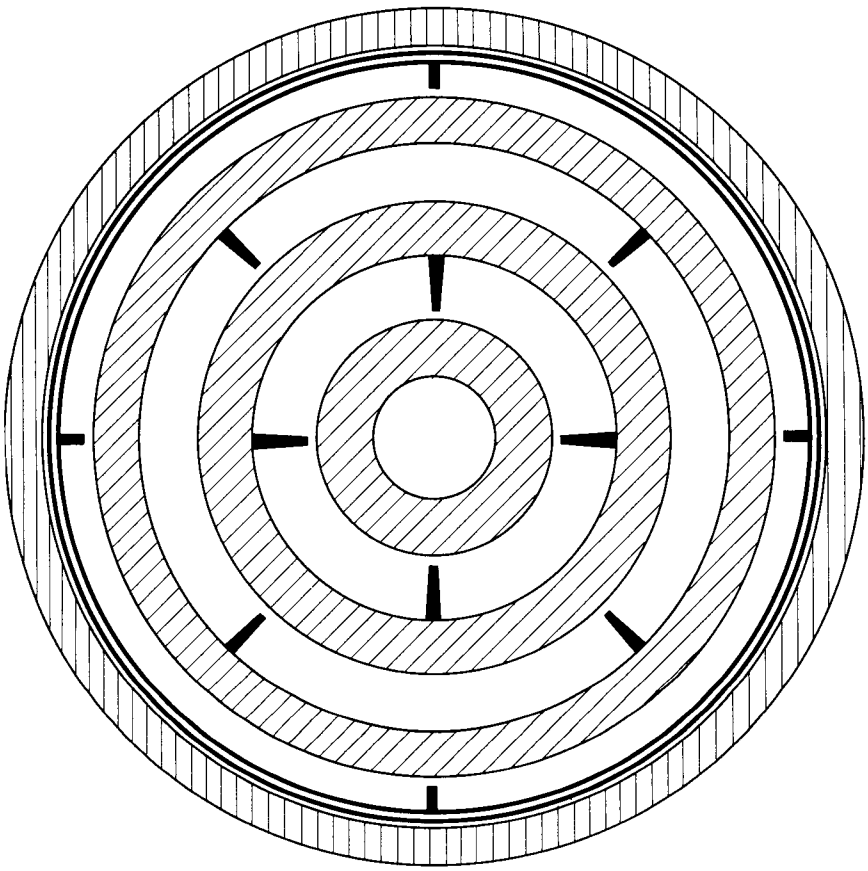
	DIMENSIONS (in.)	
	OD	ID
PRESSURE TUBE	4.75	4.43
OUTER LINER	4.37	4.31
INNER LINER	4.25	4.19
FUEL TUBE CLADDING		
TUBE NO. 1	3.83	3.34
TUBE NO. 2	2.68	2.06
TUBE NO. 3	1.34	0.68

CLADDING THICKNESS: 0.025 in.



Zr-2.5% Nb PRESSURE TUBE

ZIRCALOY LINERS, RIBS, AND CLADDING

ThO_2 FUEL

Fig. 3.8. Cross Section of Fuel Assembly of Pressure-Tube Heavy-Water Reactor—Thorium (HWR-Th).

no separate moderator assembly; the structural graphite portion of the fuel elements provides the required graphite, and the beryllium oxide is incorporated as a central rod in each fuel element. The helium coolant enters the bottom of the core at 720°F and leaves the top of the core at 1470°F, yielding a net thermal efficiency of 44.4% and a net electrical output of 1008 Mw.

A very compact arrangement is achieved by locating the steam generators in a ring around the reactor core and enclosing the entire reactor primary coolant system inside a steel membrane-lined prestressed-concrete pressure vessel. Steam-turbine-driven helium circulators are located within the concrete pressure vessel, and the gas flow inside the pressure vessel is directed by thin-walled ducts.

Principal features of the reactor are listed in Table 3.1. The design and a discussion of the reasons for choosing the particular features selected are reported in ref. 15 and in the letters listed as refs. 16-18.

The reactor core and pressure vessel are shown in Fig. 3.9. The core approximates a cylinder 31 ft in diameter and 15.5 ft in height.

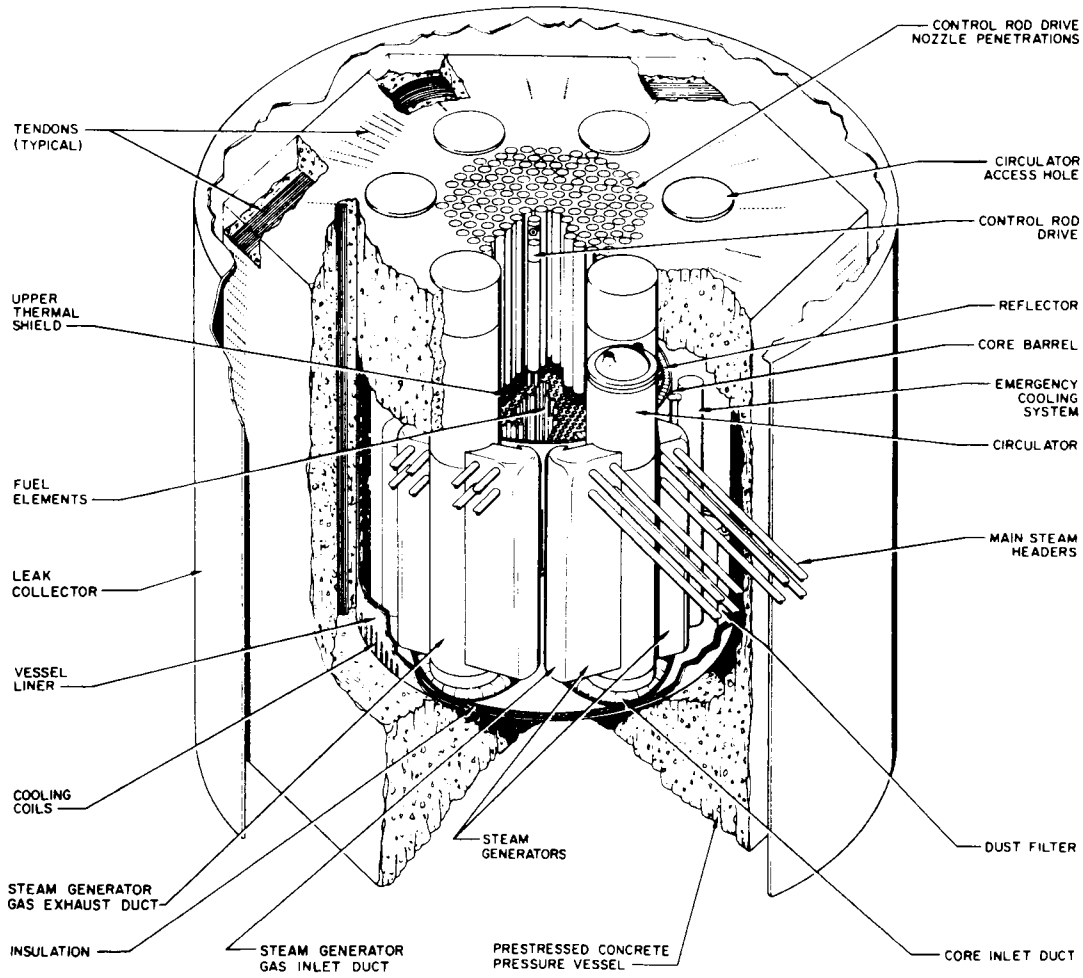


Fig. 3.9. High-Temperature Gas-Cooled Reactor (HTGR).

Surrounding the core on all sides are graphite reflectors. The basic core component is the fuel element, which is a graphite cylinder 4.5 in. in diameter and 20 ft long. Control rods of similar outward shape and containing 30 wt % boron in graphite are also located in the core, but there is no metal structure within the active core.

The reference design consists of 5489 fuel elements and 242 control rods on a 4.70-in. triangular pitch. The top and bottom reflectors are integral parts of the fuel element. All core components are supported by a steel core support plate and are located by standoff pins.

The fuel element is shown in Fig. 3.10. It consists of an annular cylinder of graphite that fits over a central cylinder of BeO. The fuel consists of packed beds of particles located in 14 holes within the graphite body in a "telephone dial" arrangement. The fuel particles are uranium and thorium carbide, 400 and 1200 μ in diameter, respectively, including a pyrolytic-carbon coating for fission-product retention. A fission-gas purge is provided by using graphite that allows a small flow of helium from the main coolant stream through it. A gap between the graphite body and the BeO spine collects the purge flow and directs it to an internal fission gas trap.

Six steam-generator—circulator loops supply steam at 3500 psi and 1050°F, with reheat to 1050°F, to a modern turbine-generator unit.

Considerable attention has been given to fuel management. Two ranges of particle sizes are used so that the fertile material (thorium) can be distinguished from the fissile material (uranium). Upon reprocessing of the fuel, the two types of particles can be separated and a portion of the uranium can be discarded or sold in order to prevent the buildup of higher isotopes (particularly U^{236}) in the recycled fuel.

Sodium-Graphite Reactor (SGR)

The 1000-Mw(e) sodium-graphite reactor[19-23] is fueled with uranium carbide, cooled by liquid sodium, and moderated and reflected by graphite. The reactor is rated at 2336 Mw(th) and 1019 Mw(e). The net plant efficiency is 43.6%.

A sodium heat transfer system removes the heat generated in the core and delivers it to the steam generators. This is accomplished by circulating liquid sodium in five parallel circuits using free-surface centrifugal pumps. Each of the circuits consists of a radioactive primary loop, which transfers heat from the reactor to two intermediate heat exchangers, and a secondary nonradioactive sodium loop, which delivers the heat from the intermediate heat exchangers to the steam generators. The primary and secondary sodium loops are shown schematically in Fig. 3.11.

The power plant complex consists of a reactor building, a steam-generator structure, a turbine-generator building, and other operating and service facilities. The reactor is housed in a steel-lined concrete cavity in the operating floor of the reactor building. The cavity is capped with the loading face shield, which forms a continuation of the operating floor and provides radiation protection for operating personnel.

Beginning at the perimeter of the concrete cavity and moving radially inward, the reactor structure consists of: (1) the cavity liner, (2) the

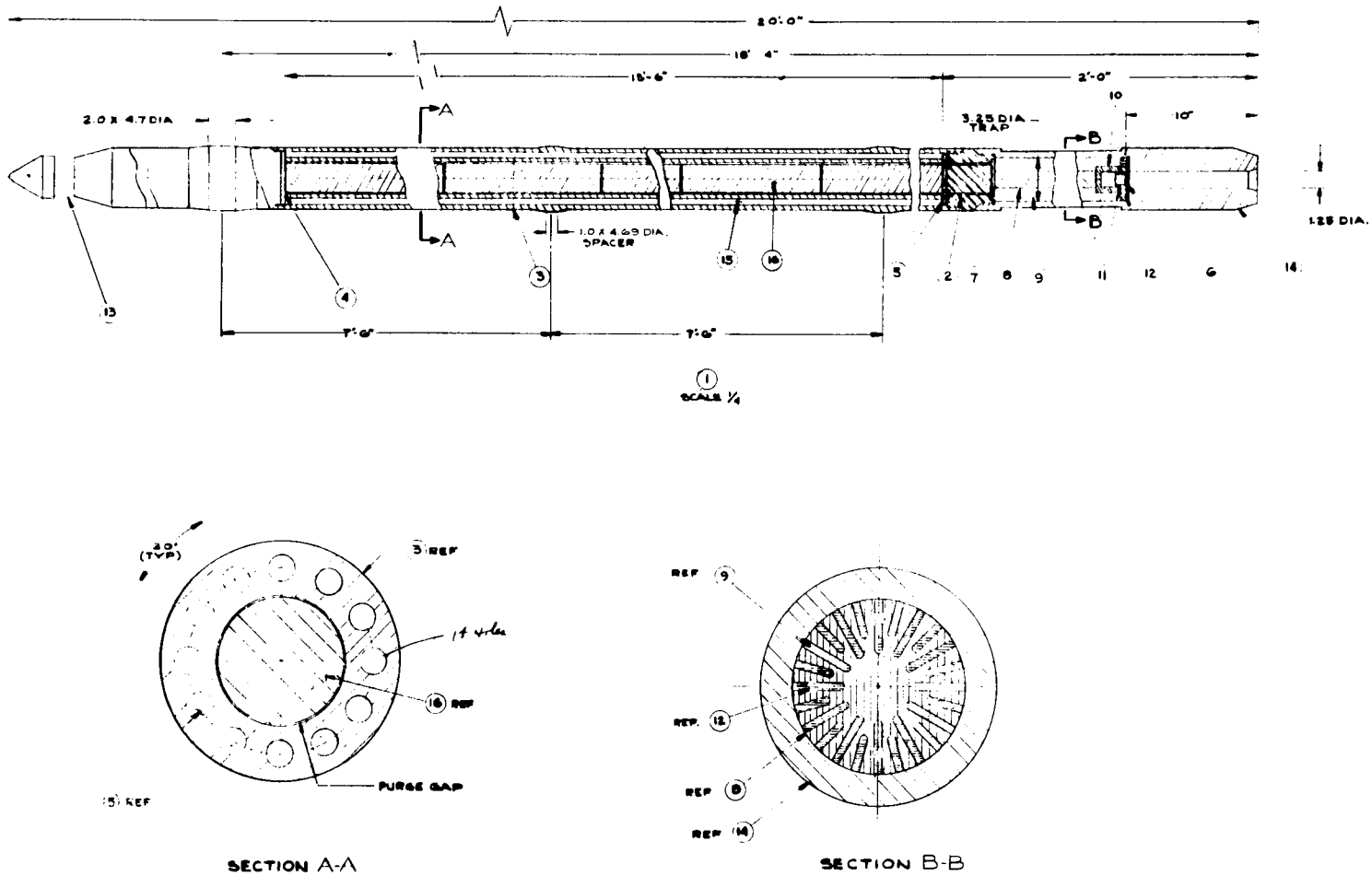


Fig. 3.10. Fuel Element of High-Temperature Gas-Cooled Reactor (HTGR).

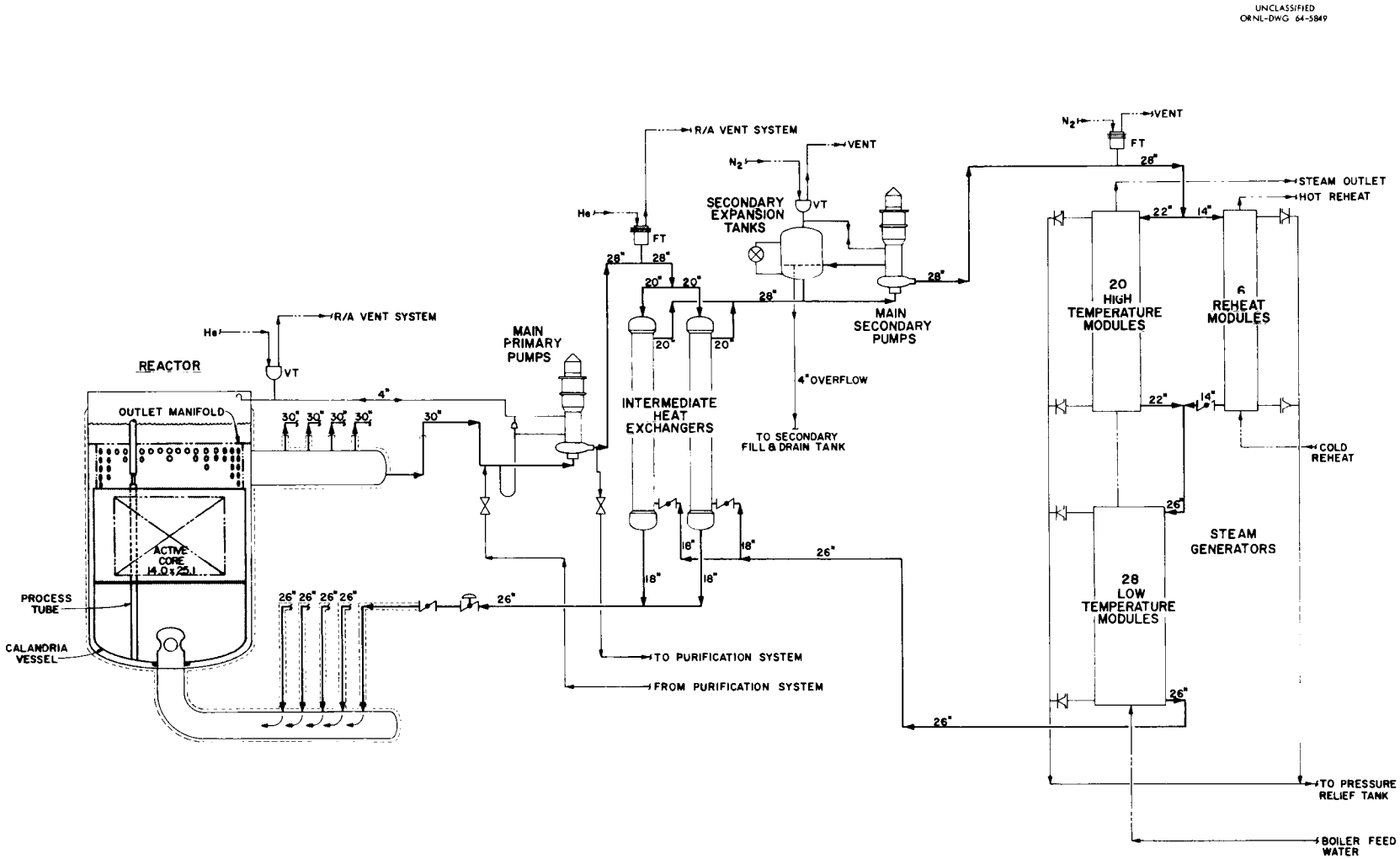


Fig. 3.11. Schematic Flow Diagram of 1000-Mw(e) Sodium-Graphite Reactor (SGR).

thermal shield, (3) the reactor vessel and its external liner, and (4) the calandria and other core components.

The vessel contains the reflector and moderator graphite and is penetrated by stainless steel process channels that provide a flow path through the core for the coolant. The fuel elements and the control rods are located within the process channels.

The active core contains 594 fuel elements and 102 control rods and is in the form of a right cylinder 25.1 ft in diameter and 14 ft high. The sodium coolant enters the reactor at 750°F, flows upward through the process tubes, and exits at 1150°F. The nominal coolant flow rate is 66×10^6 lb/hr. The maximum operating pressure in the reactor vessel is the static head of sodium, which is about 16 psig.

The fuel element assembly, shown in Fig. 3.12, is composed of 18 fuel rods and a central support tube. The outer 12 rods of the fuel

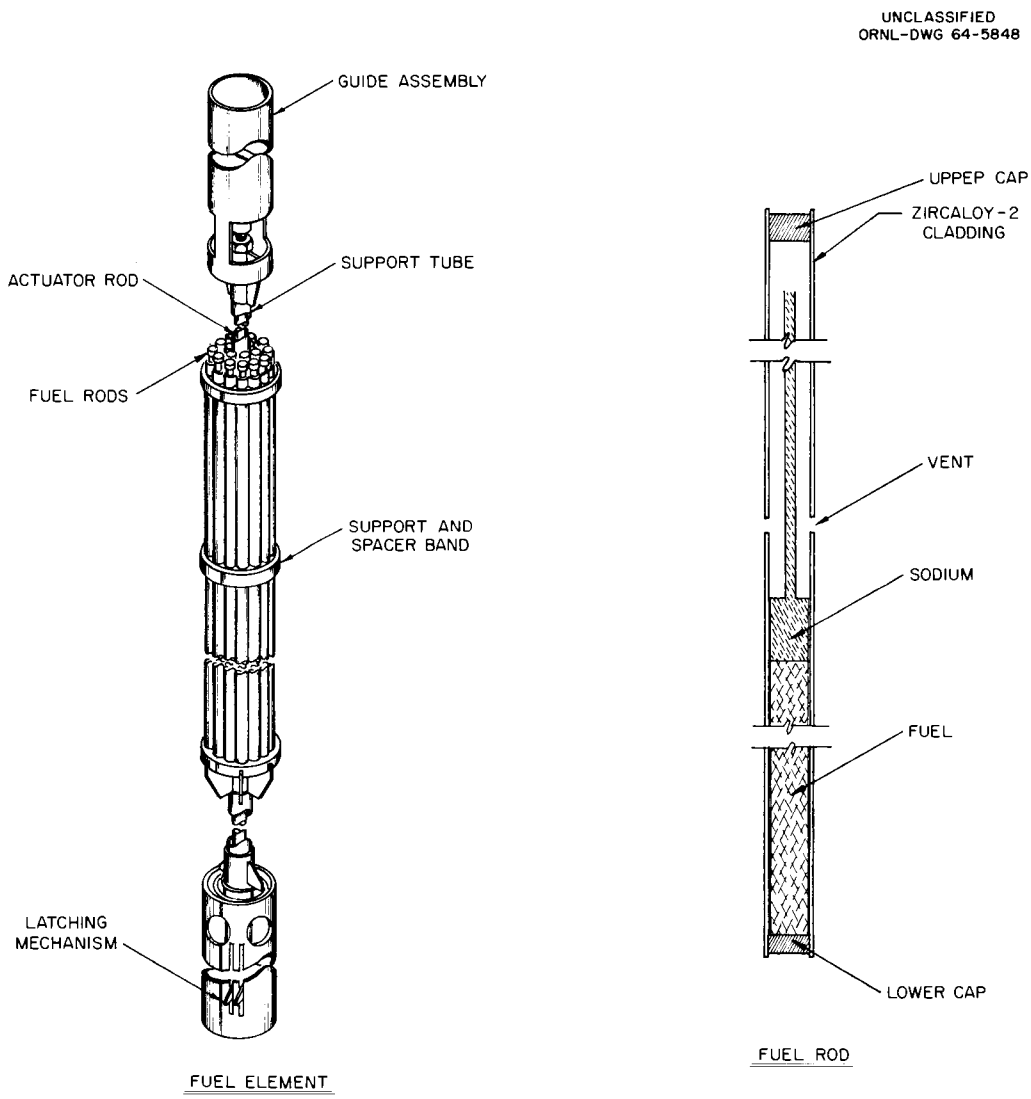


Fig. 3.12. Sodium-Graphite Reactor Fuel Element Details.

assembly are equally spaced on a 2.90-in.-diam bolt circle, and the inner 6 rods are spaced equally on a 1.5-in.-diam bolt circle. The fuel rods are made by assembling 1/2-in.-diam UC slugs into Zircaloy-2 tubes. The fuel is bonded to the tube by a 0.025-in. sodium-filled annulus to improve heat transfer. The design of the fuel rod allows the released fission gas to be vented directly to the primary coolant.[22,23]

The fuel management program calls for reloading one-sixth of the core every 7.6 months. At this time about 50 of the peripheral fuel elements will be relocated near the center of the core.

The peak-to-average power ratio is 2.25. The average fuel exposure given in the reference design is 35,000 Mwd/MT of uranium.

The steam-generator system is composed of three types of modular units: (1) low-temperature (evaporator) modules, (2) high-temperature (superheater) modules, and (3) high-temperature reheat modules. The modules are of a once-through design with sodium on the shell side and water on the tube side.

All large components were designed to eliminate requirements for massive machining operations and to achieve simplified and flexible field and shop erection and fabrication procedures. Further details of the design are listed in Tables 3.1 and 11.20.

References

PWR

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SSCR

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4. Personal communication, D. R. Hostetler, Babcock & Wilcox Co., to L. L. Bennett, April 3, 1964.

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7. Minutes of Meeting with Babcock & Wilcox Co. to discuss core design of 1000-Mw(e) spectral-shift-control reactor, meeting held January 30, 1964, R. C. Olson, Oak Ridge National Laboratory.

8. Letter, R. A. Webb, Babcock & Wilcox Co., to M. W. Rosenthal, March 17, 1964.

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HWR

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11. Letter from J. W. Wade, Du Pont, to M. W. Rosenthal, Jan. 9, 1964.

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13. Letter from C. P. Ross, Du Pont, to M. W. Rosenthal, May 1, 1964.

14. Letter from J. W. Wade to M. W. Rosenthal, May 11, 1964.

HTGR

15. General Atomic, Design Study Report for Target - A 1000 MWE High-Temperature Gas Cooled Reactor, USAEC Report GA-4706, March 9, 1964.

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SGR

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21. Letter from S. Siegel, Atomics International, to J. A. Lane, June 16, 1964.

22. F. J. Leitz (Ed.), Sodium Cooled Reactors Program, Fast Ceramic Reactor Development Program, Sixth Quarterly Report, January-March, 1963, USAEC Report GEAP-4214, General Electric Company, April 1963.

23. General Electric Company, Sodium-Cooled Reactors Program, Fast Ceramic Reactor Development Program, Ninth Quarterly Report, October-December, 1963, USAEC Report GEAP-4480, January 1964.

4. ENGINEERING EVALUATION OF REACTOR CORES*

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At the outset of this evaluation study, the proponent of each concept was requested to submit a tentative core design for a 1000-Mw(e) reactor that was attractive based on the ground rules of the study. In particular, the conditions specified were to be the most extreme contemplated conditions of temperature, heat flux, fuel exposure, etc., that were to be used in the final design.

The purpose in requesting this information was to determine whether comparable assumptions were being made by the various reactor designers with regard to thermal limitations and fuel element performance requirements. For new concepts or for concepts requiring demonstration of limiting design parameters, assumptions were considered acceptable if trends of existing experimental information indicated that the chances of demonstrating questionable performance requirements within the required time schedule appeared favorable. Inconsistencies in safety margins allowed or unacceptable performance requirements based on technological uncertainties were to be called to the attention of the proponent involved so that appropriate modifications of the core design could be made before details of the reactor concept were submitted for comparative evaluation. None of the conditions specified were altered as a result of this evaluation. However, some cases in which the operating conditions were near or possibly somewhat over the acceptable limits were pointed out.

The following discussion summarizes the results of the evaluation studies of the core thermal performance and the fuel element performance.

CORE THERMAL PERFORMANCE EVALUATION

The procedure used in the core thermal analysis consisted of the following steps:

1. Checking the flow, thermal conditions, and core physical description for internal consistency,
2. Checking DNB ratios,** heat ratings, and fuel element temperatures that might limit the reactor power output,
3. Comparing the limiting operating variables of the reactors at the rated hot channel conditions,

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*Advice and assistance in the evaluation of fuel performance was given by D. A. Douglas, Jr., J. L. Scott, and J. R. Weir.

**The DNB ratio is the ratio of the heat flux at which "departure from nucleate boiling" would be expected to occur to the heat flux at the rated overpower condition. It is also referred to as the "burnout" ratio or "dryout" ratio.

4. Comparing the thermal parameters of the reactors to identify the design differences and their relative effects on core performance requirements.

All the calculations were performed for steady-state conditions at the designer's rated overpower, and no evaluation was made of transient conditions. Axial, radial, and local factors supplied by the designers were used in analyzing thermal conditions. The details of the thermal designs and a summary of the checks performed are given in ref. 1.

Water-Cooled Reactors

Comparison of Design Features

The important thermal design characteristics are listed in Table 4.1. The values in the table are based on the design information presented by each of the proponents but, with the exception of the radial and axial power distributions, were checked by ORNL. In particular, the DNB ratios were checked using WCAP-1997, rev. (ref. 2) and other correlations and found to be consistent with the present knowledge of DNB behavior.

For all reactors except the PWR, the DNB ratio reported is the ratio of the heat flux at DNB to that at rated overpower, computed with the enthalpy rise kept fixed at that for the overpower condition. Because there is boiling in the hottest channel of the PWR, the DNB ratio for it is the ratio of the enthalpy rise at DNB to that at the rated overpower (the heat flux does not enter directly into the WCAP-1997 correlation (ref. 2) that was used). In all cases, it was assumed that the flow rate did not change from the design condition. The coolant flow rates listed in Table 4.1 are those for heat removal only and do not include fluid that bypasses the fuel elements.

The following comments are made to clarify how the designers obtained the performance data listed.

Pressurized-Water Reactor. — The PWR has a maximum-to-average heat flux ratio of 3.33, and the coolant outlet temperature of 598°F is the highest of the water-cooled reactors. Chemical shim control is used to limit the power distortion caused by the presence of control rods, but no use is made of varying fuel enrichment to flatten the power distribution. However, fuel rods having smaller diameters than normal rods are placed next to the water gap to reduce the heat rating. Short control rod followers that extend 3.7 ft into the 11 ft core and other mixing devices being developed force interchange of coolant between and within assemblies. These are assumed in the reference design to yield an enthalpy rise factor of 1.8, but achievement of this value awaits further development.

Spectral-Shift-Control Reactor. — The SSCR core is a one-pass system with flow from bottom to top of the reactor. The core is characterized by a high power density, 130 kw/liter, maximum-to-average heat flux of 1.9, and a coolant enthalpy rise factor of 1.46. These factors are largely obtained by two methods: (1) variation of the H_2O-D_2O ratio in the coolant-moderator for shim control and (2) axial and radial variations in fissile material concentrations to obtain a high degree of power flattening. The

Table 4.1. Comparison of Thermal Design Parameters for Water-Cooled Advanced-Converter Reactors

	PWR	SSCR	HWR-U	HWR-Th
Primary coolant heat removal, Mw	3320	3266	3600	3600
Core inlet pressure, psia	2050	2200	1900	1874
Coolant flow rate (for heat removal only), lb/hr	144×10^6	136×10^6	138×10^6	138×10^6
Maximum cladding temperature at rated power	Saturation			
Core coolant inlet temperature, °F	546	535	507	507
Core coolant outlet temperature, °F	598	594	579	579
Maximum hot channel outlet temperature at rated overpower, °F	639[a]	631	612	612
Coolant velocity in core, ft/sec	12.7	19.4	43.6	30
Core pressure drop, psi	34	22	43	24
Number of coolant loops	7	5	6	6
Ratio of maximum to average heat flux in core	3.33	1.9	1.69	1.69
Maximum heat flux at rated power, $Btu/ft^2 \cdot hr$	544,000	470,000	820,000	621,000
Ratio of maximum to average enthalpy rise	1.8	1.46	1.15	1.15[b]
Flow reduction factor used	1.07	1.08	1.02	1.02
Fluid mixing factor used	0.94	0.95	1.00	1.0
Maximum $\int k\, d\theta$ of fuel element (hot assembly, 10% overpower), w/cm	45.8	49	43	32[b]
Minimum DNB ratio at rated overpower	1.23 (boiling)	1.28	1.75	1.75
Rated overpower factor	1.15	1.2	1.1	1.1
Core power density, kw(t)/liter	80.7	130	21.4	16.05
Core length, ft	11.0	10.9	15.0	15.0
Core diameter, ft	12.8	10.17	22.4	25.83
Number of fuel assemblies	264	156	516	688
Moderator temperature, °F	572	565	140	140
Fuel management	3-batch scatter reloading	Zoned enrichments, single-batch reloading	3-batch scatter reloading (power-flattened zone)	On-line refueling (power-flattened zone)

[a]Bulk boiling.

[b]Equilibrium core.

moderate value of the peak heat flux of 470,000 Btu/ft^2·hr at rated power is a result of the low value of the peak-to-average heat flux ratio.

Heavy-Water Reactor—Uranium. — The HWR-U design takes advantage of the fact that the coolant and moderator are in separate loops by forcing the coolant through fuel tubes at a high velocity (43.6 ft/sec) and operating with a high surface heat flux. Low values of peak-to-average power density are obtained by using axial reflectors and ganged control rods for radial flux flattening. The reactor uses in-pile instrumentation and measures flow and temperature rise on each pressure tube.

The pressure tubes are orificed to obtain an equal temperature rise in each fuel assembly. In addition, the three concentric fuel tubes are spaced and sized so that an equal temperature rise and burnout safety factor is achieved in each of the flow channels through an element. The 72°F coolant axial temperature rise, which is the highest of the water-cooled reactors, helps reduce the D_2O inventory in the circulating system. The outlet temperature of approximately 580°F is moderate relative to the other reactors.

Heavy-Water Reactor—Thorium. — The HWR-Th is designed with the same considerations as the HWR-U. However, the number of pressure tubes is increased from 516 to 688, and the D_2O coolant velocity consequently was decreased from 43.6 to 30 ft/sec. The coolant channels were altered slightly in size to give the same enthalpy rise and radial temperature differences as the HWR-U, and thus the safety margin for both reactors is about the same. The fuel tubes of the thorium reactor have a lower heat rating than those of the uranium reactor and, therefore, are more conservatively designed.

Conclusions

All the water-cooled reactors sponsored should prove adequate from a thermal design viewpoint. While it would be desirable to rate the reactor designs on the basis of "design conservatism," this is most difficult. The PWR appears to have the least conservative design but also has the most backup in operating experience of similar reactors. The SSCR has an all-around balanced thermal design based on the current limits, provided, however, that the axial and radial peaking factors can be held within the specified range. Both of the heavy-water systems seem to have adequate thermal designs. Although the thorium reactor has a lower power density to reduce protactinium losses, low density is also required by a high ripple factor occurring early in the reactor life. Further optimization of the power density of the thorium reactor might benefit the design.

Design changes to allow for any of the considerations mentioned above should have relatively small effects on the total power costs. All the water-cooled reactor thermal designs were therefore accepted without change.

High-Temperature Gas-Cooled Reactor

A thermal analysis of the HTGR core was made based on the data supplied by General Atomic. For this analysis the thermal conductivity of the bed of fuel particles was taken to be 1.5 Btu/hr·ft·°F, except around

the periphery of each fuel column, where a value of 1.1 Btu/hr•ft•°F was used. The graphite thermal conductivity was assumed to be 14 Btu/hr•ft•°F to allow for the decrease caused by irradiation.

A "pie" section of a fuel element containing centers of adjacent fuel columns was taken to be a section of symmetry, with one fuel column being exactly opposite a column in an adjacent fuel element. It was assumed that all heat generation occurred in the fuel particles, and the surface between the BeO and graphite was taken as an adiabatic surface. A mean helium heat transfer coefficient was calculated using the usual procedure, and peripheral variation from the mean was calculated using the data of Palmer and Swanson.[3] Neglecting axial conduction, the heat-transfer section of symmetry was transformed to a rectangular slab, and the appropriate corrections were made. The resulting two-dimensional conduction problem was solved on an IBM-7090 computer using the relaxation technique.

Temperatures were calculated using transverse maximum-to-mean heat generation ratios of 1.0, 1.3, and 1.4 with the specified axial heat flux distribution. For all cases, the maximum fuel temperature occurs at one-fourth the way through the core. The calculated temperature profile through the hot fuel point one-fourth the way through the core for a maximum-to-mean transverse power generation ratio of 1.3 is shown in Fig. 4.1. Because of the relatively high thermal conductivity of graphite, the effect of this peripheral variation on fuel temperatures was small.

Although several simplifying assumptions were made, the results of this analysis are believed to be slightly conservative. No temperature alleviation by axial conduction was considered, and all heat generation was assumed to take place in the fuel itself (where the thermal conductivity is the smallest). The maximum temperatures obtained were in reasonable agreement with values previously given by General Atomic.[4]

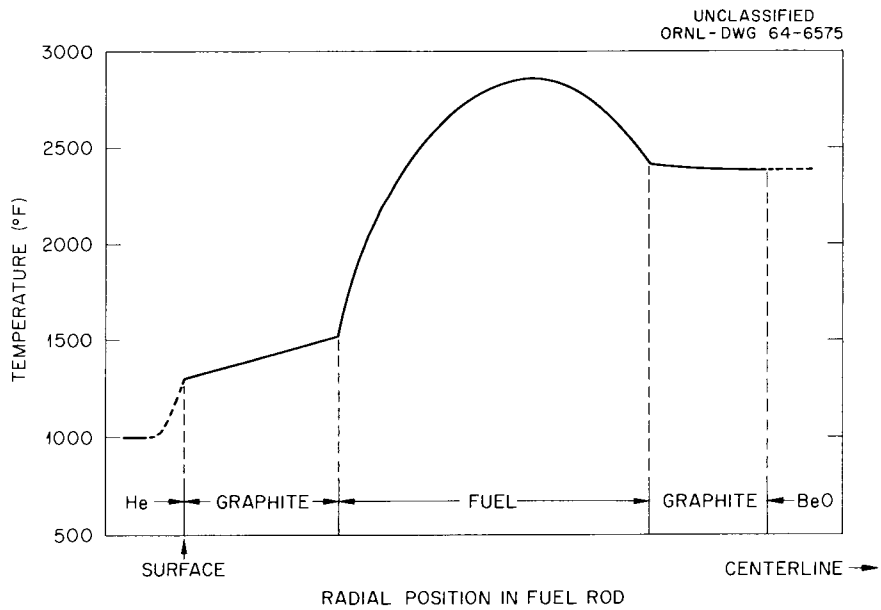

Fig. 4.1. Fuel Rod Maximum Temperature Profile One-Fourth the Way Through the Core; Maximum-to-Mean Transverse Power Generation = 1.3.

Precise knowledge of the effective thermal conductivity of the particular fuel is of prime importance in establishing the maximum fuel temperature. Although values determined experimentally by General Atomic were used in this analysis, confirmation of the thermal conductivities of coated particles of the specific size and distribution proposed for the fuel element is necessary.

A simplified analysis showed that slight misalignments of the fuel rods would not cause significant changes in the maximum fuel temperature due to the relatively high thermal conductivity of the graphite. The use of surface roughness to improve local heat transfer could be beneficial, but it should be noted that surface roughness would only affect the relatively small film temperature difference.

Sodium-Graphite Reactor

A thermal analysis of the 1000-Mw(e) SGR was made based on information supplied by Atomics International,[5] with the exception that the cladding material was taken to be 0.020-in.-thick Zircaloy-2 (resulting in a fuel pin outside diameter of 0.590 in.) and the process tube inside diameter was taken as 3.95 in. The bolt circle diameters of the inner and outer fuel pin rings were taken to be 1.50 and 2.90 in., respectively. Forced-convection heat transfer coefficients were estimated using the data of Maresca and Dwyer.[6]

In lieu of better information, some simplifying, but probably conservative assumptions, were made. Considering an entire process tube, it was assumed that the overall peak-to-average power ratio of 2.25 occurred at a longitudinal position 1.12 ft downstream of the midpoint of the reactor core. The radial peak-to-average power ratio of 1.50 was assumed to exist throughout the entire length of this process tube. On this basis, the maximum fuel temperature was calculated to be 2030°F, as compared with 2000°F reported by Atomics International. At this position the mean sodium temperature was ~1100°F and the temperature drops were approximately as follows: maximum fuel center to surface, 670°F; across sodium gap, 50°F; across Zircaloy-2 wall, 160°F; across sodium coolant "film," 50°F.

FUEL ELEMENT PERFORMANCE EVALUATION

The fuel element performance of each reactor was analyzed to determine the extent to which it conformed to current fuel technology and to identify areas of concern that might be expected to limit the life of the fuel element. All the reactors represent some advance in fuel technology. For a fuel element design to be considered successful, the incidence of failure must be low for the life of the core and the design operating conditions. Failures may occur with any fuel element when an improbable set of factors occur in combination, and almost any fuel element design can be made to survive with a low incidence of failure by lowering such factors as burnup and power ratings. The performance characteristics of a fuel element are not reliably established by a few successful experiments or by those in which only one or two of the several important factors are tested. Rather, the determination of reliability must await

the statistical accounting afforded by a multitude of tests that simulate operational conditions. For some of the proposed reactors, test data and operating experience are sufficient to predict that the objectives can be obtained. For others, considerably more testing will be required before the design is proven and the operational limitations are established.

Fuel element characteristics for the different reactors are listed in Table 4.2. The fuel lifetimes and burnups shown are those provided by the sponsors and do not reflect the reoptimization of the fuel cycles in this study. Heat ratings listed are the maximum values that will exist at rated power. Higher ratings will occur locally at overpower or hot-channel conditions, but such values will seldom be encountered and should have little effect on performance. The performance of the different fuel elements is analyzed individually in the following sections.

PWR Fuel Element

The fuel element of the PWR is 155 in. long, with 132 in. fueled. It consists of 194 tubes 0.378 in. in diameter and 40 smaller tubes 0.320 in. in diameter. These tubes have 0.020-in.-thick Zircaloy-4 walls and are filled with UO_2 pellets of 96.5% theoretical density. They are designed to operate for three years at a linear heat rating of 16 kw/ft and a maximum heat flux of 544,000 Btu/hr·ft^2 to a maximum burnup of 30,000 Mwd/MT of uranium. The peak cladding temperature is 640°F. The coolant is pressurized water at 2050 psi.

Fuel rods of this type (UO_2, Zircaloy-4 clad) were run as the core loading in the Shippingport PWR to a burnup of 10,000 Mwd/MT of uranium at peak surface temperatures of about 580°F with complete success. Some elements held over from the first core loading have been operated there to 15,740 Mwd/MT of uranium successfully, and satisfactory performance is projected to 23,610 Mwd/MT (three normal core lifetimes) at this location. This test is continuing.[7]

Other reactors (Dresden, CVTR, etc.)[8] have also operated with this type of fuel element under similar service conditions to burnups of 10,000 Mwd/MT of uranium, routinely, and to higher burnups on occasion. These tests also show satisfactory performance of the element.

High-burnup tests of UO_2 have been performed with plate-type elements containing thick wafers of fuel.[9] These tests indicate slight swelling of 96%-dense UO_2 at a burnup of 54,000 Mwd/MT of uranium. This result is apparently independent of fuel temperature from 1340 to 3840°F. Stainless steel-clad UO_2 has also been run to a burnup of 27,000 Mwd/MT of uranium in Core II of the Yankee Reactor without significant trouble.

The use of vibratorily compacted UO_2 in the PWR is more of an extrapolation of current technology at the heat ratings and burnup proposed. The possibility of waterlogging failure or failures resulting from possible contamination of the fuel during fabrication increase the tendency to failure beyond that which may be expected from pelletized fuel. It appears likely that vibratorily compacted fuel of approximately 89% density would operate satisfactorily, but experimental verification is required.

Table 4.2. Fuel Element Characteristics of Advanced-Converter Reactors

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Fuel element configuration	Square array, 16 × 16 rods	238-rod bundle	Three concentric tubes	Three concentric tubes	Graphite cylinder	Rod cluster, Na annulus
Active core length, ft	11.0	10.92	15.0	15.0	15.5	14
Fuel rod or tube diameter, in.	0.378 0.320	0.420	Up to 3.630	Up to 3.830	4.5	0.590
Fuel life, years	3	2.1	1	1.7	6	2.68
Fuel	UO_2	UO_2-ThO_2	UO_2	ThO_2-UO_2	ThC_2-UC_2	UC
Fuel fabrication	Pelletized	Vibratory compaction	Vibratory compaction and swaging	Vibratory compaction and swaging	Coated particles	Cast, 0.50 in. in diameter
Fuel theoretical density, %	96.5	88	92	92		
Maximum $\int k\, d\theta$ at rated power	41.2 35	39.5	34	25	2900°F	2000°F
Average burnup in maximum power rod, Mwd/MT	30,000	45,000	19,100	27,600	80–90 at. % U 6–10 at. % Th	40,000
Core average burnup, Mwd/MT	25,000	30,000	15,400	22,200		35,000
Cladding material	Zircaloy-4	Zircaloy-4	Zircaloy-2	Zircaloy-2	Pyrolytic carbon coating	Zircaloy-2
Cladding thickness, in.	0.020	0.025	0.025	0.025		0.020
Maximum cladding surface temperature, °F	640	650	626	626		1250
Maximum cladding surface heat flux at rated power, Btu/hr·ft^2	544,000	470,000	820,000	621,000	140,000	760,000
Coolant	H_2O	H_2O-D_2O	D_2O	D_2O	He	Na
Coolant inlet pressure, psia	2050	2200	1900	1874	450	48
Coolant inlet temperature, °F	546	535	507	507	720	750
Coolant outlet temperature, °F	598	594	579	579	1470	1150

Hydrogen absorption of the Zircaloy-4 cladding does not appear to be a major problem. The data of Kass[10] suggest that the hydrogen concentrations will not exceed about 150 ppm during the proposed three-year life.

Thus the PWR fuel element appears to be a very modest extrapolation of present technology, and it is expected that it would perform satisfactorily.

SSCR Fuel Element

The fuel element of the spectral-shift-control reactor is a 10.9-ft-long assemblage of Zircaloy-4 tubes 0.42 in. in diameter and 0.0255 in. thick. The tubes are filled with UO_2-ThO_2 that is vibratorily packed to 88% of theoretical density and will operate to average burnups of 30,000 Mwd/MT with a peak burnup of 45,000 Mwd/MT. The maximum nominal heat flux is 470,000 Btu/hr·ft^2, equivalent to $\int k\, d\theta = 39.5$ w/cm. Under conditions of 10% overpower and maximum hot channel factors, the heat rating will be 49 w/cm, a condition which will seldom be approached. The maximum cladding surface temperature is 650°F, and the coolant is pressurized water at 2200 psi, with inlet and outlet temperatures of 535 and 594°F. The residence time of fuel elements is 2.1 years.

Technical Review

The heat ratings of rod-type fuel elements for water-cooled reactors are limited so that the fuel will not exceed a specified operating temperature. The limiting heat rating depends on the thermal conductivity of the fuel, which, in turn, is dependent upon fabrication method and density. The effective thermal conductivity is somewhat lower for vibratorily packed fuels than for sintered fuels. Although limits on heat rating for vibratorily compacted fuel elements have not been established, experimental results indicate that the performance may be comparable with that of a similarly dense sintered fuel.

Canadian reactors are designed on the basis of a power rating of $\int k\, d\theta = 40$ w/cm (16 kw/ft linear heat rating), but available information suggests that this may be increased to 45 w/cm. These values are based on 95%-dense UO_2 pellets with an oxygen-to-uranium ratio of 2.005. The newer reactors in this country (Dresden and CVTR)[8] operate with similar or slightly greater heat ratings. Considerable testing has been done at higher heat ratings, and satisfactory performance is generally obtained. Recent information indicates that design specifications for UO_2 fuels will be revised upward to allow incipient melting in the overpower condition.[11]

At ORNL, packed-powder fuel rods prepared from sol-gel ThO_2-UO_2 have been irradiated and examined for burnups ranging from 4000 to 45,000 Mwd/MT at heat ratings somewhat under SSCR design ratings and at higher heat ratings for shorter exposures. These data have indicated that sintering and grain growth structures were produced at peak values of heat rating some 10 w/cm higher than for comparable UO_2 specimens.[12]

Vibratorily compacted UO_2-PuO_2 irradiated in the PRTR[13] gave satisfactory performance at low burnup. Experiments for measuring the fission-gas release of approximately 90%-dense specimens indicated that gas release increased linearly with heat rating and was less than 20% for $\int k\, d\theta = 40$ w/cm.

Areas of Concern

The principal uncertainty for the SSCR fuel elements is that vibratorily compacted fuel elements have not been tested at design heat ratings and to the burnup proposed. Although available information indicates that the fuel elements will operate as proposed, difficulty might be caused by (1) the generation of pressure from released fission gases, (2) the problem of adsorption or occlusion of gases on the fuel during fabrication, (3) hydriding of the Zircaloy-4 cladding, and (4) waterlogging.

Release of Fission Gases. Calculations indicate that, at the maximum exposures, fission-gas pressures would not exceed the coolant pressures at 30% of fission-gas release. With the reactor operating, pressures at 100% fission-gas release for the average exposures of 30,000 Mwd/MT would develop stresses in the Zircaloy-2 cladding of only 20,000 psi. Thus, fission-gas release does not appear to be a major problem.

Occluded Gases. A considerable problem in connection with the use of vibratory compaction of fuels is that of adsorption or occlusion of gases or other impurities during fabrication. Adsorption of water or contamination with oil from processing equipment might cause internal hydriding of the sheath, and occluded gases might contribute to pressures from fission-gas release.[13] Successful fuel element performance will depend on adequate control of these factors.

Hydrogen Pickup. Available data on hydrogen pickup of Zircaloy-4 at the maximum surface temperature of 650°F suggest that hydrogen concentrations from external corrosion will not exceed approximately 90 ppm during the operating life of 2.1 years.[10]

Waterlogging failure may be encountered with 88% dense fuel rods fabricated by vibratory compaction. Such a failure has been reported by Eichenberg et al.[14] for a pellet rod containing 80% dense hot sintered pellets. Occasional failures have been noted in defected test rods with nonsintered fuel, although others have operated without difficulty. Extensive studies by Bettis and GE-APED are in agreement that waterlogging is not a problem in fuel rods containing pressed and sintered UO_2 pellets with densities in the range of 92 to 96% of theoretical. As additional experience is accumulated with vibratorily compacted fuels, the significance of waterlogging as a failure mechanism will be determined.

Conclusions

These data suggest that the fuel element for the SSCR is designed with a modest extrapolation of current technology, and it is believed that it would operate satisfactorily at design conditions. The results of experimental tests presently in progress at ORNL will permit more definite conclusions in the near future.

HWR Fuel Element

The fuel elements for the pressure-tube heavy-water reactors consist of three vertical concentric tubes 15 ft long that are contained in Zr–2.5% Nb alloy pressure tubes. The fuel tubes, which are 0.025-in.-thick Zircaloy-2, are filled with oxide by vibratory compaction and swaged to 92% fuel density. The coolant is D_2O, with inlet and outlet temperatures of 507 and 579°F. The maximum fuel cladding temperature is 626°F.

Two similar reactors have been designed, one fueled with slightly enriched UO_2 and the other fueled with ThO_2-UO_2. The dimensions of the fuel tubes differ slightly, as shown in Chapter 3. The fuel management scheme for the UO_2 reactor is such that the nominal heat rating in the flat zone will be $\int k\, d\theta = 39$ w/cm during the initial one-third of its life but will drop to 29 w/cm during the latter one-third of its life. The overall average exposure in the flat zone will be 15,400 Mwd/MT, with a maximum exposure of 19,100 Mwd/MT. The fuel element life is 1 year.

The reactor fueled with ThO_2-UO_2 would operate at a lower heat rating of $\int k\, d\theta = 25$ w/cm, which would exist for its entire life. The maximum exposure is 27,600 Mwd/MT, and the life of the fuel element is 1.7 years.

Technical Review

The ThO_2 fuel element has not yet been tested, but promising results have been obtained from tests on single annular elements containing UO_2 fuel with Zircaloy cladding on both the outer and inner surfaces.[15] Tubes 2.1 in. OD with 0.030-in.-thick cladding have been tested at $\int k\, d\theta = 25$ w/cm to 12,800 Mwd/MT and at 29 w/cm to 10,000 Mwd/MT. Tubes 2.5 in. OD have been tested to 1000 Mwd/MT. Tubes as large as the outer fuel tubes have not yet been tested, and nested tubes have not been tested as a unit. Du Pont believes that the data indicate that UO_2-fueled tubes should operate satisfactorily at maximum heat ratings of $\int k\, d\theta = 30$ to 35 w/cm to exposures of at least 20,000 Mwd/MT. Experiments at ORNL with irradiated ThO_2-UO_2 rods suggest that physical changes which develop during irradiation will occur at heat ratings of about 10 w/cm higher than with UO_2 under similar conditions.[12]

Areas of Concern

Several areas of concern foreseen for the performance of these fuel elements are discussed briefly below.

Concentric Tube Concept. The concentric tube design has not yet been tested at heat ratings and burnups of interest. Welding of spacing ribs to the cladding surface may have detrimental effects on mechanical properties or corrosion resistance of the cladding during the life of the fuel element. The possibility of wear or fretting corrosion damage to the cladding of the adjacent tube at its contact with the end of the spacer must also be considered. Such damage would be expected to be an increasing function of fuel exposure, since pressures at contact points would increase were there any tendency for the fuel tube to bow during operation.

Tests on the nested tube element are also necessary to establish that coolant flow to different channels of an element can be controlled to provide adequate cooling at all surfaces. Nonuniform heating on opposite halves of a fuel tube, which might result from differences in fuel thickness, would result in unequal expansion and tend to bow the tube. Within limits imposed by the spacers, such bowing might reduce external coolant channel spacing at the heated region, with the possible effect of reduced cooling and further temperature increase at the hot region.

Similarly, if internal and external sheaths of a tube operated at different temperatures, the differences in expansion would stress the sheaths of long tubes. However, it does not seem likely that the temperatures would differ enough for this to be a major problem.

Collapse of Zircaloy-2 Sheaths. The design of the heavy-water reactors is predicated on the use of large-diameter thin-walled Zircaloy-2 fuel sheaths. Under the design conditions the sheaths would collapse if they were not restrained by the fuel itself. Thus, the cladding is nonfreestanding and, in this respect, is unlike the cladding on fuels for the other reactors being evaluated in this study.

The use of nonfreestanding cladding, together with an annular fuel design, could lead to serious problems if the fuel were of low density and the heat rating were high. When UO_2 operates above about 1800°C in a temperature gradient, voids migrate by a process of sublimation of UO_2 on the hot side of the void and condensation on the cooler side. The void migration is toward the maximum temperature and results in a central void in rod-type fuel elements. Since the resulting configuration is stable with rods that have freestanding cladding, this void migration has not been cause for alarm. With annular shapes cooled on both the exterior and interior, void migration would produce a central void that would not be geometrically stable under operating forces. The fuel would collapse into the void and leave a gap between the fuel and the outer sheath. Since the cladding is not freestanding, it would also collapse into the void. However, it would do so by forming a ridge. In the ridge itself the plastic strain would be high and cracking would occur and cause fuel element failure. Thus, it is imperative to keep a balance between the heat rating and density to prevent excessive fuel shrinkage.

Failures of the above type were observed at Du Pont on UO_2 annular elements with low fuel densities (82 to 87% theoretical density) and at high heat ratings (40 to 50 w/cm). In subsequent tests, 2.1-in.-OD swaged tubes of 89 to 90% theoretical density have not failed at heat ratings of 30 w/cm and exposures of up to 13,000 Mwd/MT using 0.030-in.-thick Zircaloy-2 cladding.[15] Thus, fuel densification and cladding wrinkling should be no problem in the reactor fueled with ThO_2-UO_2, since sol-gel ThO_2-UO_2 is more refractory than bulk UO_2 and may be operated at about 10 w/cm higher heat rating before comparable structural changes occur.[12] The higher heat ratings proposed for the UO_2 reactor would result in more densification of the fuel, and additional experimental testing is required to establish that collapse would not occur at the heat ratings and burnup proposed.

The inner sheath of the fuel tube would have little strength to resist pressures from gases released by the fuel, and the critical pressure for collapse would decrease rapidly with increased inner sheath diameter.

During reactor operation, when high fuel temperatures would cause greatest internal pressures, collapse would not occur unless the internal pressure exceeded the coolant pressure of about 1900 psi. Du Pont data suggest that an oxide-filled tube of 90% density, operating at 30 w/cm, would release about 20% of the fission gas formed during irradiation. The internal pressure would not exceed 900 psi if all the fuel were operated at a 1600°C central temperature to 20,000-Mwd/MT burnup (with allowance made for 50 cm^3/kg of UO_2 of gas initially present or desorbed during irradiation).[15] The actual pressure would be less because the axial flux gradient would reduce the average temperature and burnup and because there is a 9-in. void provided at the end of the fuel element.

It would appear that a more critical condition for collapse of inner fuel sheaths would exist when the reactor was shut down and depressurized. Internal pressure for the above conditions at a temperature of 100°C would be about 275 psi. The collapse pressure for the unsupported inner sheath of the larger fuel tube is less than 30 psi. The fuel offers some support against collapse by preventing the inner sheath from assuming an oval shape. Du Pont results suggest that this effect may increase resistance to collapse by at least a factor of 3 to 6.[16]

Occluded Gases. Adsorbed or occluded gases or other impurities introduced during fabrication have been a considerable problem with vibratorily compacted fuels. Du Pont experienced internal hydriding and excessive internal pressures during irradiation of early fuel tubes, but, as shown in recent tests, improved fabrication techniques have avoided these difficulties. Hanford has irradiated some 4500 individual rods to low burnup in the PRTR, and of an estimated 32 failures, 24 were attributed to the presence of contaminants in the fuel.[13] Rigid control of fabrication techniques apparently minimizes failure.

Hydrogen Pickup. Based on the data of Kass,[10] hydriding of the Zircaloy-2 cladding from corrosion by the coolant does not appear to be a problem at proposed conditions.

Conclusions

No difficulty is foreseen with ThO_2-UO_2 fuel at proposed burnup and operating conditions of this reactor. The more severe conditions proposed for the UO_2 fuel are much less conservative, and it must be established that the proposed heat ratings are suitable for use with the larger annular tubes. However, additional experimental work is necessary before either fuel element design can be considered proven technology. Available data on smaller tubes appear promising, but none of these tubes have yet been examined after design burnup. Since geometric considerations indicate that larger tubes will be more sensitive to conditions likely to cause failure, research is necessary to determine that the larger tubes will operate at design conditions. Research is also needed to demonstrate that the three concentric tubes can operate as a unit to the proposed burnup. Potential problems are possible detrimental effects of spacers on performance of fuel cladding and possible effects of bowing or distortion of fuel tubes. Du Pont is currently engaged in research to develop the nested fuel element, and tests now under way should provide additional information on performance.

HTGR Fuel Element

The HTGR reference fuel element[4] is an extruded or machined graphite cylindrical shape 20 ft long by about 4.5 in. in diameter that stands vertically in the reactor. The fuel, in the form of loose pyrolytic-carbon-coated particles of ThC_2 and UC_2, is contained in fourteen 0.5-in.-diam holes with centers equally spaced about 0.5 in. from the periphery of the cylinder. A 2 1/2-in.-diam BeO moderator spine is located at the center of the fuel cylinder. The active length of the fuel and moderator spine is 15 1/2 ft. The helium coolant, which enters the reactor at 450 psi and 720°F and leaves at 1470°F, flows upward in the channels between the fuel elements.

The HTGR fuel element does not rely on a metal cladding to retain fission products and is thus called an "unclad" fuel element. Two provisions are made to prevent contamination of the primary coolant with fission products. First, the individual fuel particles have pyrolytic-carbon coatings; 200-μ carbide particles with 100-μ coatings have been shown to retain fission products very well to burnups in excess of 20 at. % heavy metal at a temperature of 1370°C. Second, a helium purge is provided to each element by means of a manifold at the bottom of the element. The manifold operates at a reduced pressure so that helium flows through the pores of the graphite, past the coated fuel particles, down the annulus between the graphite element and the BeO spine, and out the manifold. Fission-product traps are provided both within the fuel element and in an external cleanup system to remove fission products from the purge gas. The purified helium is then returned to the primary coolant stream.

The maximum fuel temperature is given as about 1550°C. The maximum graphite temperature will be 1400°C. The maximum operating temperature of the BeO is 1430°C. The entire fuel element will be replaced as a unit at the rate of one-sixth of the elements each year, so the intended operating life is about 6 years.

The fuel is an intimate mixture of two types of coated particles, which are contained as loose coated particles in the 1/2-in.-diam holes spaced near the periphery of the graphite cylinder. Each fuel element, as charged, will contain about 14.1 kg of thorium as 1000-μ-diam particles of ThC_2 coated with about 100 μ of pyrolytic carbon, and about 0.52 kg of enriched uranium as 200-μ-diam particles of UC_2; the UC_2 particles are also coated with 100 μ of pyrolytic carbon. During the 6-year term in the reactor, the ThC_2 will achieve about 5 to 10 at. % burnup of heavy metal and the UC_2 particles about 80 to 90 at. % burnup.

Technical Review

The fuel element has not been tested, and many of the significant details of the design have not been worked out. There is, however, extensive coated particle technology, and the proposed design represents a reasonable extrapolation of this technology. There exists extensive information on the effects of irradiation on graphite of many different types, and the irradiation behavior of BeO is well developed. However, a number of features of the element are sufficiently novel that a considerable amount of research and development, particularly irradiation

testing, will be required before this system can be considered as existing technology. Furthermore, there is great flexibility in the system, and relatively minor changes may result in a design that should be satisfactory even though it might not be optimum.

Areas of Concern

Analysis of the reference design suggests several problems that must be resolved before it can be established that the fuel element will operate as intended. These are

1. Selection of optimum fuel element configurations,
2. Establishment of performance of coated-particle fuels at design conditions,
3. Determination of compatibility and radiation damage to the BeO moderator spines,
4. Determination of effects of failure of the graphite body holding the tiny fuel particles.

These subjects are discussed separately below.

Fuel Configuration Problem. A problem area in optimizing the design of the fuel element is recognized by General Atomic and is a principal reason that fuel element design has not been fixed. Basically, the problem involves limiting the maximum operating temperature while achieving the desired fuel loading in the available fuel volume with particles that have coatings of sufficient thickness to contain the fission gases during the required fuel life. This necessitates that the ratio of fuel to coating be much higher (larger particle diameter) than has yet been tested. The alternate solution of providing greater volume for fuel does not appear to be feasible without extensive design changes.

Performance of Fuel Particles. Satisfactory performance of coated-particle fuels has been demonstrated in experimental tests up to a maximum burnup of 24% at maximum temperatures of 1370°C.[17] These data were obtained on UC_2 particles fabricated by General Atomic that had triplex coatings 82 μ thick on 180-μ-diam cores. No coating failures were observed, and the data indicate that fission gases were retained. Similar data obtained at lower burnup and temperature have also indicated satisfactory performance of these and other types of coated particles.

The knowledge of coated-particle performance has not advanced to the point where complete retention of fission gases can be assumed; therefore, a purge stream is provided to remove fission products released from defective particles. It is proposed that the UC_2, which comprises less than 4% of the fuel, can be heavily protected with pyrolytic carbon coatings so that few particle failures will occur even at 80 to 90 at. % burnup. This proposal is reasonable, but the thickness of the coating material required to prevent particle failure must be determined.

A simplified theoretical analysis has been used to predict dimensions of coated particles that may allow incorporation of the desired fuel loadings in the available fuel volume. This analysis is based on provision of a void volume as a porous inner layer (about 50% dense) of the coating to accommodate swelling of the fuel core and to allow expansion of released fission gases so that internal pressures do not cause fracture of the outer

impervious coatings. Using somewhat optimistic assumptions, the results suggest that the 200-μ-diam UC_2 particles, operating to 80 at. % burnup, would require a porous layer 74 μ thick, with 112 μ of impervious outer coating. The ThC_2 particles, operating to about 6 to 10 at. % burnup, would require about 1000-μ-diam cores with a 50-μ porous layer and a 100-μ-thick outer coating. These values should be recognized as preliminary rough estimates of fuel-particle configurations that might be able to accommodate fuel core swelling and the pressures of released fission gases and which could be loaded into the fuel volume of the reference design. The technology of coating particles as large as 1000 μ in diameter and for applying relatively thick inner coatings of high porosity has not yet been demonstrated at the production rates and economic levels projected in this study, but it is believed that such processes can be developed.

Even if coatings remain intact, some species of fission products have penetrated sound coatings. While gases are retained well, other long-lived radioactive species, such as barium and cesium, will migrate through the coatings. To date this migration has been found to be a problem above 1600°C, but current tests are of short duration and the HTGR fuel element is to operate for 6 years. Even if the barium migrated through the coating, it would probably be adsorbed on the graphite and not released to the coolant. Cesium, on the other hand, would probably be removed by the helium purge, but tests are needed to clear up the present uncertainties.

In addition to fission-product migration, fuel migration through the coatings may be serious during the long-time operation at high temperatures. Such migration is time and temperature dependent and may be affected by other variables as well, such as the kind of coating and existence of thermal gradients. Tests have shown significant penetration in furnace tests of 3000 hr at 1700°C and no penetration during 10,000 hr at 1550°C.[18]

Performance of Moderator Spines. Irradiation damage to BeO arises principally from a volume increase as a result of anisotropic expansion of individual crystals that eventually leads to grain-boundary separation and powdering of the BeO. Damage is proportional to the integrated fast-flux exposure and decreases with increasing temperature; however, severe fracturing developed in several specimens at 1100°C at exposures exceeding 2×10^{21} neutrons/cm^2 ($E > 1.0$ Mev). Volume increase at 650°C increased linearly from 2% at 0.8×10^{21} neutrons/cm^2 to about 6% at 2.2×10^{21} neutrons/cm^2 (ref. 19). Specimens with fine grains (<5 μ) and with certain additions have exhibited superior resistance to radiation damage, and the possibility of exploiting these features should be investigated.

The available data indicate that swelling and cracking of the moderator spines is likely to develop during the 6-year cycle. Since the BeO is not a structural member, it would appear that damage will not be serious if sufficient clearance is provided to ensure that the graphite fuel elements are not stressed excessively.

BeO and graphite will react to form BeC_2 and CO at temperatures exceeding 1400°C, but available data indicate that the rate of reaction is slow and dependent on maintaining intimate contact. Johnson[20] heated dense BeO and graphite and found no reaction below 2300°C. Other experimenters, using loosely compacted powders, found appreciable reaction at 1800°C.[21] It appears that experimental data should be obtained, but it is likely that compatibility between BeO and the graphite is not a serious matter.

Performance of Graphite Bodies. Experimental evidence is needed to establish that the graphite fuel elements will remain dimensionally stable. Failure of the graphite bodies may allow highly active loose particles to escape to the coolant stream. The possible breakage of a fuel element during removal from the reactor, with attendant loss of the loose particles into the reactor, would appear to be a significant hazard, and some thought should be given to the cleanup costs that might be involved. An alternate solution to the problem would be to fabricate fueled compacts with graphite in the interstices between the coated particles, as proposed in alternate designs. These compacts would show improved heat transfer and thus lower central temperatures; however, fabrication costs and reprocessing costs would probably be increased.

The current HTGR design calls for a homogeneous mixture of coated particles of two different sizes. To achieve the required homogeneity may be quite difficult. If segregation occurred, those regions containing the higher fissile contents would be excessively hot and might cause the fuel elements to break as a result of thermal stresses. It would also be difficult to inspect the fuel elements to establish that there was the desired homogeneity in each element.

Conclusions

The HTGR element is based on considerable extrapolation of current technology in both fabrication and performance of coated-particle fuels. Achievement of objectives will require development of techniques for producing and coating large particles at reasonable costs and extensive testing to determine the effects of operating temperatures and coated particle design parameters on the life and burnup achievable.

From the aspect of fuel performance, it would appear that some advantages might accrue if the ThC_2 and UC_2 were mixed and fabricated as homogeneous particles so that average particle burnup would be within the limits of burnup presently tolerable. This approach would only slightly reduce the maximum size of the particles, however. A similar effect might be achieved by diluting the UC_2 with ZrC. A reduction of maximum operating temperature and residence time in the reactor would, of course, reduce the probability of coated particle failure and would also affect the economics.

Several other modifications of the HTGR fuel element have been proposed, including the use of oxide particles instead of carbides. An economic analysis, presented in Chapter 8, illustrates the anticipated savings from use of oxide particles. A number of other design changes are under study at General Atomic, and it appears likely that the reference design studied in this analysis will not be the final one.

SGR Fuel Element

The fuel element for the sodium-graphite reactor is an 18-rod cluster of fuel tubes around a single central tube that supports the fuel rods and spacers. The fuel rods are 17 1/2 ft long and contain 14 ft of 0.5-in.-diam hyperstoichiometric (4.8 to 5.1% C) UC slugs assembled in

0.020-in.-wall Zircaloy-2 tubes. A 0.025-in.-thick annulus containing sodium is used to improve heat transfer and accommodate swelling of the UC. It is proposed to vent released fission gases to the sodium coolant by use of a design proposed and tested by General Electric[22] (see Fig. 3.12).

Based on the original design with 0.010-in.-thick stainless steel tubes, the maximum hot-spot central fuel temperature was estimated to be 2000°F, with a maximum nominal value of 1870°F. The maximum inner cladding temperature was 1250°F. Substitution of 0.020-in.-thick Zircaloy-2 for the thinner stainless steel will probably increase these temperatures by about 60°F.

Significant data on design parameters of the fuel element are listed in Table 4.2.

Areas of Concern

Several potential problem areas are listed and discussed briefly below.

1. The vented fuel element concept has not yet been proved for fuel elements using a sodium bond. The problems of filling the capsule and protecting the sodium from the atmosphere after loading would involve more expense than for conventional fuel elements. The possibility of vent blockage leading to cladding rupture seems remote but must be considered. The proposed design would seem to preclude interchange between sodium in the annulus and the coolant and suggests that all fission products except rare gases will remain within the sodium in the annulus. A vented oxide element without a sodium bond that was irradiated at $\int k\, d\theta = 82$ to 93 w/cm to 14,500 Mwd/MT released 50% of the fission gases generated through a vent into static sodium.[23] Results of analysis of the sodium indicate that release of activity to the sodium coolant in the form of solid daughter products of rare gases is likely to be small.

2. A major problem with the use of UC is the control of the carbon content. If the carbon content is below 4.8%, some uranium metal will be present at grain boundaries. This uranium results in increased fission-product release and increased fuel swelling. If, on the other hand, the carbon content is greater than 4.8%, UC_2 platelets will be present in the microstructure. The sodium readily transports the excess carbon to the cladding. Although BMI has reported control of the carbon content to ±0.03 wt %, many fabricators question that such control is economical. With present Atomics International practice, a range in carbon content of 4.8 to 5.1 wt % C would be expected. It would be desirable to improve the control of carbon content on a production basis, and AI is investigating means for doing this.

3. A possible major problem in the use of Zircaloy-2 cladding will be pickup of oxygen, which will exist as an impurity in the sodium coolant. Oxygen reacts readily with zirconium at 1250°F, but little information on the limits or kinetics of the reaction is available. Atomics International reports that the oxygen level in moderator cans in the SRE did not increase significantly during operation at temperatures up to 1050°F. They believe

that hot and cold traps will maintain the oxygen level in the sodium coolant at 6 to 15 ppm. Significant oxygen pickup has been reported in zirconium in sodium containing 15 to 20 ppm O_2, the value being equivalent to about 0.4% in 0.020-in.-thick zirconium exposed 190 hr at 1250°F.[24] Atomics International is well aware of the problem, which is being extensively investigated, because the SRE Power Expansion Program will use Zircaloy-2 moderator cans in 1200°F sodium.

Similarly, the interior of the cladding will be exposed to carburization from the sodium bond. Atomics International reports that zirconium alloys are highly resistant to carburization at 1200°F.[25] However, positive proof that this will not impair the operating characteristics of such elements will be obtained only after a substantial number of elements have been operated under prototype reactor conditions.

4. Little information exists on the performance of UC fuels to an average burnup of 35,000 Mwd/MT at the maximum operating temperature of 2000°F, and none exists at the maximum burnup proposed. Data obtained by United Nuclear Corporation indicate that hyperstoichiometric UC swells less and releases less fission gases than material with less carbon.[26] Data are given in Table 4.3. Similar data have been obtained at lower burnup by Atomics International. Extensive cracking of the carbide occurs, but it apparently has no deleterious effects. Although additional data are needed to verify the performance of UC at design conditions, available data suggest that swelling of the UC can be expected to be within the limits allowable by the sodium bond.

Table 4.3. Results of Irradiation Experiment[a] on Cast UC

	Capsule No. 1		Capsule No. 2	
	Specimen 1	Specimen 2	Specimen 1	Specimen 2
Carbon content of UC, %	5.20	4.70	5.20	4.70
Burnup, Mwd/MT	36,000	34,800	22,900	22,900
Diameter increase, %	5.7	8.9	(b)	(b)
Diameter increase relative to burnup (at. %), %	1.2	1.9		
Density decrease, %	(c)	(c)	6.6	6.2
Density decrease relative to burnup (at. %), %			2.1	2.0
Fission heat generation, kw/ft	29	29	35	36
Average operating temperature, °F	1830	1880	2080	2050
Maximum operating temperature, °F	2100	2100	2270	(d)

[a]Data from ref. 26.

[b]Fuel too badly broken for diameter measurements to be made.

[c]Reliable measurements could not be made.

[d]Thermocouple failed before fuel achieved 1 at. % burnup.

5. The original design of the SGR was based on use of 0.010-in.-thick stainless steel tubes with hypostoichiometric UC (4.5 to 4.8% C) at a maximum fuel temperature of 2000°F. Swelling and gas release of UC increase as the carbon content decreases, but performance of UC within the carbon range specified appears satisfactory on the basis of modest extrapolations of the limited data available. AI is currently investigating the performance of UC at the temperatures of interest. A potential problem of carburization of the stainless steel cladding exists. Data obtained at 1200°F suggest that carburization is not severe with hypostoichiometric UC for times up to one-third the expected life of the fuel.[25] Therefore, it appears that the stainless steel-clad fuel element would operate satisfactorily, but the life cannot yet be predicted.

Conclusions

Operation of the sodium-graphite reactor fuel elements as proposed would appear to depend upon satisfactory performance of the vented fuel element design and upon the ability to maintain low impurity levels in the sodium coolant to prevent embrittlement of the Zircaloy-2 cladding. Although available information on performance of UC at proposed conditions is promising, further research is needed to demonstrate that performance is satisfactory up to the maximum burnup anticipated.

Additional research to verify performance in several areas is therefore necessary before it can be determined that the fuel elements will operate as proposed. Such research is in progress at Atomics International and is necessary to exploit the advantages of UC as a reactor fuel. Some advances beyond current technology are to be expected. It appears likely that successful operation of the fuel element will be achievable, possibly with alternate clad materials, or by slightly relaxing the maximum burnup or temperature requirements.

SUMMARY AND CONCLUSIONS

The core designs submitted for the advanced-converter reactor evaluation fall into two categories from the viewpoint of design conservatism. These categories are (1) reactors designed on the basis of current technology and (2) reactors designed on the basis of extrapolations well beyond existing data.

The spectral-shift-control and pressurized-water reactors are included in the first category — those based on current technology — with the reservations that the power peaking factors used in the thermal analyses have not been checked by physics calculations and the enthalpy rise factor of 1.8 claimed for the PWR has not yet been demonstrated. Both reactors use heat ratings of fuel rods that are within current technology. The SSCR operates at a slightly higher heat rating than the PWR, and the PWR is less conservative than the other water-cooled reactors on heat transfer and burnout considerations.

The second category includes the heavy-water reactors, the high-temperature gas-cooled reactor, and the sodium-graphite reactor. Although

heat transfer considerations appear to be adequate, assumptions regarding fuel element performance for these reactor concepts are well beyond demonstrated technology.

For the heavy-water reactors, extensive development and testing will be required to demonstrate that the nested-tube concept is satisfactory at the proposed operating conditions, and the success of this design is likely to remain uncertain until this is done. Potential problems are possible failures of the larger fuel tubes by collapse or wrinkling of the cladding and possible damage to the cladding by the spacers of adjacent tubes. The low heat rating proposed for the ThO_2-UO_2 fuel, and the greater stability of the fuel itself, suggest that the design of the thorium-loaded reactor may be the more conservative. The longer proposed life of the ThO_2-loaded fuel element, however, will make it more susceptible to fretting damage of the cladding.

The sodium-graphite reactor is based on extrapolation of the technology developed through operating experience with the SRE and the Hallam reactor. Further research will be necessary to demonstrate successful performance of the vented fuel concept, the ability to maintain sufficiently low impurity levels in the sodium coolant to prevent corrosion of the Zircaloy cladding at the higher operating temperature, and satisfactory performance of the UC fuel at the temperature and burnup proposed. These problems are being actively investigated, and the Power Expansion Program for the SRE will furnish additional information on the ability of the proposed fuel element to operate at design conditions.

The fuel element for the high-temperature gas-cooled reactor is based on considerable extrapolation of coated-particle technology. Since no reactors have operated with similar fuel elements, extensive development and testing will be necessary to demonstrate performance of the reference fuel element. The design of the reactor has great flexibility, and several modifications of the fuel element have been proposed. However, continued testing will be required before the potential life of the fuel elements can be predicted.

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5. ENGINEERING REVIEW OF PLANT DESIGN FEATURES

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In order to evaluate the technical feasibility and capital costs of a reactor concept, it is necessary to have detailed information on the design of the major components of the plant, particularly those components that are especially for a given concept and whose construction may require development or extrapolation of existing technology. With this in mind, each sponsor was requested to supply descriptions of the systems of his plant giving functional requirements, design specifications, and installed costs of the major equipment items. In most cases, however, design studies on the 1000-Mw(e) reactor plants had not progressed to the point where complete details could be supplied. Consequently, the amount and completeness of information of this type varies from one concept to another.

The report* on the PWR design, which was used in this study as a standard for comparison, contains the most complete description of the design requirements. However, Westinghouse has indicated that changes brought about by further investigations since the time the PWR report was written would significantly affect the design and costs given. Adjustments were made in this study for some of the apparent changes, such as modification of pressure vessel requirements due to code changes, but time did not permit investigating the full effect of other possible design modifications recommended by Westinghouse.

The SSCR equipment requirements were also taken to be the same as those of the PWR, except for differences in reactor primary equipment determined from information supplied by Babcock & Wilcox.

Du Pont furnished general descriptions of some of the principle HWR systems and sizes of major equipment items for which cost breakdowns were given. They indicated, however, that design optimization studies for the 1000-Mw(e) reactors had not been completed.

The reports on the HTGR and SGR concepts provided enough description to identify most of the requirements and criteria for the major components of those plants. However, neither report had sufficient detailed drawings or other information from which a complete and independent evaluation of feasibility or cost could be made.

In view of the limited amount of detailed information available, it was necessary to restrict the engineering evaluation to a comparison of the design features of systems controlling the major portion of the plant capital costs. The procedure used for design normalization consisted of (1) identifying systems of a plant with functional requirements in common with those of another plant, (2) making adjustments in equipment specifications to permit cost comparisons on a more equivalent basis, and (3) reviewing features of equipment peculiar to a given reactor type with regard to feasibility and the status of technology.

Many of the features of the water-cooled plants, such as those of the turbine-generator systems, containment structures, coolant systems,

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*See references listed in Chapter 3.

auxiliaries, and even some of the reactor vessel equipment, can be compared directly and normalized to a reference design (in most cases the PWR). The supercritical steam systems of the HTGR and SGR also lend themselves to direct comparison with each other. However, most of the other systems of these latter two plants are not directly comparable with each other or the water-cooled plants. In these cases, other information on equipment of the type proposed was needed for comment on the feasibility of the concept.

In addition to providing a basis for capital cost comparisons, the design concepts were reviewed to compare possible operational and maintenance problems. The feasibility of a particular design is dependent on its ability to satisfy the prescribed plant availability requirement. Potential problems of this type are discussed in this chapter to indicate where comprehensive evaluation of operation and maintenance procedures should be made when more specific information becomes available.

Comparison and Discussion of Principal Design Considerations

Reactor Vessels and Internal Structures

Several reactor vessel types are represented by the reactor concepts included in this study. The basic differences in approach to vessel design are due primarily to the type of coolant used. The water-cooled reactors all require high pressures (of the order of 2000 psia) to suppress vaporization of the coolant. The gas-cooled HTGR requires a moderately high pressure of 450 psia to improve the heat removal characteristics of the coolant. The SGR is sodium cooled and requires little or no pressure above atmospheric conditions. Some of the pertinent features of the reactor vessel designs are compared in Table 5.1.

The PWR and SSCR cores are sufficiently compact for the designers to have enclosed the entire core inside a single pressure vessel. However, these vessels are quite large compared with those based on the present state of the art, particularly with regard to wall thickness. Thus the feasibility of the reactor designs is closely identified with the feasibility of fabricating the proposed high-pressure vessels to nuclear vessel quality requirements. Manufacturers of large vessels are confident that they can fabricate vessels of the size and quality needed for these applications, but development work in vessel fabrication will be required to establish the feasibility firmly.

The relatively large cores of heavy-water reactors make it very unlikely that conventional pressure vessels could be utilized in their design. However, use of individual pressure tubes for each fuel channel avoids the problem. This solution is readily adaptable to the heterogeneous nature of the heavy-water reactor.

The HTGR and the SGR, which operate at higher temperatures, have very little in common with the water reactors. The maximum coolant temperatures in the water reactors are well below the 650°F that the ASME Unfired Pressure Vessel Code permits for carbon steel vessels without requiring

Table 5.1. Comparison of Reactor Vessels

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Description	Cylinder with hemispherical heads	Cylinder with dished heads	Cylindrical tank	Cylindrical tank	Cylinder with flat heads	Cylinder with flat heads
Material	Carbon steel	Carbon steel	Stainless steel with Zr-2.5% Nb pressure tubes	Stainless steel with Zr-2.5% Nb pressure tubes	Prestressed concrete	Stainless steel
Liner	Stainless steel	Stainless steel			Carbon steel	Stainless steel
Number of loops	7	5	6	6	6	5
Principal shell thickness, in.	11.2	10.75			180 (15 ft)	1.75
Inside diameter, ft	15.83	13.5			56.5	34
Inside height, ft	44	33			40	47
Estimated weight, tons	591	313				
Removable head	Yes	Yes			No	No
Insulation location	External	External			Internal	Internal and external
Operating pressure, psia	2050	2200	1700	1700	450	48
Design pressure, psia	2500	2500	2000	2000		
Design temperature, °F	650	650	628	628		
Reactor inlet temperature, °F	546	535	507	507	720	750
Reactor outlet temperature, °F	603	594	579	579	1470	1150
Equivalent core diameter, ft	12.8	10.17	22.4	25.85	31.1	25.1
Active core length, ft	11.0	10.92	15.00	15.00	15.5	14.0

reduction in the allowable design stresses, whereas the nominal core inlet and outlet temperatures for the HTGR are 720 and 1470°F, with corresponding temperatures of 750 and 1150°F for the SGR. Both the HTGR and SGR have graphite moderators that result in core sizes comparable with those of the heavy-water reactors. Since the SGR is sodium cooled and does not require a high-pressure vessel, the relatively high operating temperatures do not present a serious limitation in the design of the vessel. The HTGR utilizes a prestressed-concrete reactor vessel to contain the coolant pressure, but its entire interior must be insulated to maintain the wall below 150°F to prevent deterioration of the concrete.

PWR. The PWR vessel is cylindrical with a hemispherical lower head and a hemispherical-closure upper head. The dimensions for the Westinghouse reference design are 16.83 ft in inside diameter by 44 ft in inside height, and the cylindrical wall is 13 7/8 in. thick. A detailed study was conducted by Combustion Engineering Corporation to determine the feasibility of building and installing a vessel of this size. They concluded that it is feasible to fabricate the vessel in the shops and transport it to the site by barge.

The information used by Combustion was supplied by Westinghouse early in the design study. Westinghouse indicated in their report that the reactor could actually be accommodated in a vessel 15.83 ft in inside diameter. For the purpose of this study it was assumed that the smaller vessel would be used in conjunction with the reduced wall thickness permitted by Section III of the ASME Unfired Pressure Vessel Code. The effect of this is to reduce the wall thickness to about 11 1/8 in., a reduction of 2 3/4 in. on the cylindrical portion of the vessel. This should materially reduce fabrication problems and result in a corresponding reduction in cost. The approximate weight of the vessel is reduced from 770 tons to about 590 tons by these changes.

Further reductions in the size, weight, and cost of this vessel would result from the combined effect of other design changes that are contemplated. These changes are (1) adoption of the rod cluster control concept in place of the cruciform control rods used in the reference design and (2) placing all the coolant nozzles in a single course around the vessel. No attempt was made to evaluate the effect of these changes on the design or cost.

The major difficulty in the design of the reactor vessel internal structures for the reference design will be the proposed flow-distribution system required in the upper portion of the core. This problem may be alleviated somewhat by the rod cluster control rod concept being considered by Westinghouse.

SSCR. The SSCR reactor vessel, although smaller, is very similar to the PWR vessel in design. The inside diameter is 13.5 ft, and the overall inside height is 33 ft. There are ten coolant nozzles located in a single course above the core to accommodate five primary coolant loops. The coolant flows downward between the core and the vessel and then upward through the core. The vessel has a nominal cylinder wall thickness of 10 3/4 in. and a weight of approximately 310 tons. Such a vessel could readily be fabricated and transported to the site.

HWR. The heavy-water reactor designs being considered do not have a reactor vessel in the ordinary sense; pressure tubes are used to contain

the coolant pressure. These tubes are individual pressure vessels that pass through the core for each fuel channel; there are 516 tubes in the HWR-U and 688 in the HWR-Th. The pressure tubes pass through a calandria vessel that contains the D_2O moderator and reflector. The design is a single-pass type with coolant connections at each end. The calandria vessel is 25 ft in diameter and 19 ft high, with flat heads. The calandria shell and heads that form a tube sheet for the Zircaloy channel liners are fabricated from stainless steel, and the pressure tubes are of Zr-2.5% Nb alloy, with transition joints to stainless steel at each end. Flexible piping arrangements connect the pressure tubes to torus-shaped headers at each end of the core. One end of the pressure tube has a mechanical closure and seal to permit removal of the fuel assemblies. In the HWR-U design the reactor is refueled from above the core while shut down. Refueling of the ThO_2-fueled reactor is on-stream, and the operations are carried out below the core. The heavy-water reactor designs require either on-stream refueling or frequent refueling shutdowns. This will necessitate very reliable mechanical closure seals and refueling equipment in order to maintain the plant availability at an acceptable level.

The pressure tubes and calandria required in this design must operate satisfactorily at high integrated neutron and gamma radiation doses. In recognition of possible difficulties, the pressure tubes have been designed as semipermanent reactor components and can be replaced if necessary. The inside walls of the pressure tubes are exposed to the hot coolant in the HWR-U design but are protected by Zircaloy liners in the HWR-Th design. The creep and strength requirements for the Zr-2.5% Nb alloy pressure tubes were investigated and found by extrapolation of existing data to be satisfactory, with the exception of the apparent ommission of a corrosion allowance for the HWR-U design. An interpretation of refs. 1-4 indicates that an allowance of up to 0.030 in. of additional wall thickness may be needed for the HWR-U pressure tubes. Extensive testing of material under simulated or actual service conditions will be required to demonstrate the adequacy of the design for a 30-year service life. Should it be necessary to replace the pressure tubes or portions of the calandria during service, a significant penalty could be imposed on the plant availability.

HTGR. The HTGR utilizes a prestressed-concrete pressure vessel, with the entire primary circulating system, consisting of the reactor core, six heat exchanger-gas circulator loops, and emergency coolant heat exchangers, contained inside it. A 3/4-in.-thick carbon steel membrane lines the vessel to provide a gastight enclosure. The vessel is essentially a right circular cylinder with flat heads; it has an inside diameter of 56.5 ft and an overall inside height of approximately 40 ft. A nominal wall thickness of 15 ft and a head thickness of about 20 ft is required by the 450-psig design pressure. This thickness is more than adequate for the wall to serve as biological shielding. The vessel must be internally insulated in conjunction with cooling coils embedded in the concrete and attached to the liner to keep the concrete temperature below 150°F. Control rod drives are located above the core, and refueling is performed through the rod drive nozzles.

The HTGR design considered in this study is a major departure from the design of the Peach Bottom Reactor and has less precedence to compare with than other reactor concepts. There are some features, however, that

can be compared with the British Oldbury Reactor Station, which is currently under construction and scheduled to go into operation in 1966. Table 5.2 shows this comparison.

As noted from Table 5.2, both reactor systems have the following prominent features in common:

1. Indirect-cycle gas cooling,
2. A prestressed-concrete reactor vessel,
3. Reactor and primary circulating system enclosed inside a single pressure vessel,
4. No complete containment around the primary vessel.

Moreover, features of the HTGR prestressed-concrete vessel are similar to those of the Oldbury design. A 1/8-scale model of the Oldbury design has been built and tested. Extensive French experience with prestressed-concrete vessel construction is also available.

Table 5.2. Comparison of HTGR and Oldbury Stations

	1000-Mw(e) HTGR	One Unit of Oldbury Station
Thermal power, Mw(t)	2270	834
Coolant	Helium	CO_2
Nominal system pressure, psia	450	365
System pressure drop, psi	8.0	12.6
Core inlet temperature, °F	720	473
Core exit temperature, °F	1470	770
Circulator type	Vertical, axial flow	Horizontal, single-stage, axial flow
Number of coolant loops	6	4
Circulator drive backup	None	Electric motor
Vessel geometry	Cylinder with flat heads	Cylinder with flat heads
Vessel inside diameter, ft	56.5	77
Vessel inside height, ft	40	60
Vessel wall thickness, ft	15	15
Vessel head thickness, ft	20	22
Vessel liner thickness, in.	0.75	0.5
Secondary containment vessel	None	None

A questionable feature of the HTGR concept is the omission of a secondary containment building to contain the coolant in the event of a failure of the primary system. The acceptance of this philosophy of containment in the United States is questionable for large power reactors constructed in the near future, even though a precedent has been set by existing British reactor plants. Adoption of this approach may be possible after actual long-term operation experience with concrete vessels is obtained.

Gas-cooled reactors have two serious problems to contend with that are not present in the liquid-cooled reactors. They are loss of circulation and loss of coolant. This is due to the inherently low heat capacity and absence of vaporization effects in gaseous coolants in conjunction with the inability to keep coolant in the core should there be a failure of the vessel. The use of the prestressed concrete pressure vessel and the "all-inside" design does reduce the probability of a loss of pressure accident, but it does not eliminate it. Basically, the inherent safety characteristics of a prestressed-concrete pressure vessel are only applicable to vessels without penetrations and discontinuities in the vessel wall. The introduction of penetrations makes them vulnerable to failures of types attributed to steel pressure vessels, since the penetrations and closures are generally the same as those encountered in steel vessels. The large number of penetrations at the top of the HTGR vessel through which control rods and fuel elements are inserted and withdrawn are therefore vulnerable to failure, even though the possibility of failure is extremely remote. Considerably more engineering and development thus is considered necessary to bring the prestressed-concrete vessel technology to the point that its full potential benefit is secured. As a result, for the cost estimate in this study the pressure-containing secondary containment, in accordance with an alternate design by General Atomic, was assumed to be required.

The leakage of steam into the core of a graphite-moderated reactor has been considered a serious problem. This appears to be particularly true for this concept due to the high graphite temperatures and the very short coolant circulating time in the compact primary circulating system. Since the steam pressure in the heat exchangers is about 3000 to 3500 psia higher than the gas pressure, a tube failure would introduce a large amount of steam or water into the gas stream. In this event, use of moisture sensors to initiate heat exchanger blowdown possibly could not prevent damage to the graphite and gas circulators; no means of isolating the affected loop is provided in the reference design. Thus, providing assurance that there will be no significant damage to fuel and equipment or loss of plant availability may be a problem in this arrangement.

SGR. The reactor vessel is essentially a double-walled tank open at the top and supported from the loading face shield above by hangers around the top of the vessel. The double wall has been provided to prevent loss of coolant should a failure of the inner vessel wall occur. The inner vessel is a stainless steel cylindrical shell with a dished lower head and is designed for the static sodium head and weight of the vessel and internal structure. It is 34 ft in inside diameter, 47 ft in inside height, and 1 3/4 in. thick. The sodium coolant enters the vessel at 720°F through a single 70-in. nozzle located in the center of the lower head and exits at 1150°F through a single 76-in. nozzle located above the core.

The reactor vessel is inside a secondary vessel or container and the space between is filled by inert gas. This container is formed by the concrete biological shield and a carbon steel liner. Carbon steel thermal shields are located between the two vessels.

The upper biological shield is high-density concrete lined with high-temperature reflective insulation, and it serves as the reactor loading face. There are separate penetrations through the loading face for each of the 680 fuel channels.

The graphite moderator and reflector are contained inside a free-standing stainless steel calandria vessel that is 32 ft in diameter by 28 ft high. An elliptical lower head that is an integral part of the calandria forms a coolant inlet plenum. Process tubes in the calandria are of stainless steel 3.95 in. in diameter and 0.025 in. thick and have a bellows at the upper end to compensate for differential thermal expansion. The bellows are brazed to the calandria heads to permit replacement of a defective tube, and this arrangement also permits replacement of graphite moderator pieces, which are designed to be removed through the process tube channels.

There are two major differences in the design of the 1000-Mw(e) SGR and the Hallam Nuclear Power Facility SGR currently in operation. These are the use of a calandria vessel instead of individual cans to contain the graphite and the change from stainless steel-clad fission-gas-retaining fuel elements to Zircaloy-clad vented fuel elements. The 1000-Mw(e) SGR report that is the primary source of information for this study describes the system and equipment required for fission-gas-retaining fuel assemblies. Some changes in the fuel-handling and storage procedures and equipment and in the design of the sodium cover-gas system will probably result from the use of vented fuel elements.

The use of a calandria vessel presents maintenance problems that have a serious effect on cost and plant availability. In view of the severe conditions and the fragile nature of the calandria design, the likelihood of a calandria failure cannot be excluded. There is no provision in the reference design for separating sections of the calandria vessel; hence, it is possible to completely flood the moderator and reflector with sodium if a failure of a process tube occurs. Location of a leak and survey of the graphite for damage, as well as actual repair or replacement of tubes and graphite, appear to be time-consuming and costly operations, particularly since it probably would be necessary to completely unload the core and drain the coolant. In conversations with AI representatives, it was learned that design improvements presently being considered should alleviate the problem of uncontrolled sodium leakage, but failed tubes would eventually have to be replaced. Considering the time required for refueling, about three to five months may be needed to make a single calandria repair. The use of the calandria concept thus involves a risk of an unscheduled shutdown of long duration.

Fuel Handling

The fuel-handling methods vary widely among the various concepts under consideration. Selection of a method is influenced for the most part by

peculiarities of the reactor type and considerations involving fuel management schemes, equipment costs, and refueling down time. The down time for refueling in most cases is a major determinant of the availability of the power plant.

To estimate the effect of refueling time on plant availability, a study was made based on the proposed fuel management schemes, refueling procedures, and exposure periods specified by the proponents. Whenever possible the time requirements were estimated from schedules obtained from the proponents. The refueling procedure for each reactor is described below and the refueling time and resulting availability factors are summarized in Table 5.3. The refueling factor quoted in the table is defined as

$$A = \frac{T}{T + t} ,$$

where T is the operating time between shutdowns and t is the refueling shutdown time.

PWR. The reactor is shut down once each year for refueling, and one-third of the 264 fuel assemblies is replaced each time. Fuel assemblies remain in the same location during their lifetime, and elements are replaced throughout the core. The reactor vessel head and the control rod drives and drive support grid are removed to gain access to the fuel, and the reactor cavity is flooded with borated water to provide shielding for the operator during refueling. Semiremote methods and special power-driven hoisting devices are used for handling the fuel. All fuel-handling operations can be carried out by direct viewing, but supplementary viewing is provided by closed-circuit television. Fuel assemblies are withdrawn from the reactor into the flooded reactor cavity and then moved through a canal and an underwater lock to the fuel storage pit.

Table 5.3. Plant Availability Factors

Reactor	Refueling Frequency[a]	Fraction of Core Changed	Number of Elements Changed	Days Shutdown to Refuel	Refueling Factor, A	Availability Assuming No Nuclear Equipment Failure
PWR	1 year	1/3	88	18[b]	0.953	0.916
SSCR	2 years	1	156	25[b]	0.967	0.929
HWR-U	4 months	1/3	172	6	0.952	0.901
HWR-Th	On stream			0	1.000	0.931
HTGR	6 months	1/12	458	10	0.947	0.901
SGR	8 months	1/6	99[c]	14	0.944	0.902

[a]Based on exposure times specified for reference designs.

[b]Estimated by ORNL; proponent supplied no data.

[c]Fifty elements also relocated.

SSCR. The reactor is shut down about every two years for refueling, at which time the entire core is changed. All fuel assemblies remain in the same location throughout their life in the core. The reactor vessel head, control rod drives, and support structure are removed for the refueling operation, and the fuel assemblies are drawn up into a shielded cask on a bridge spanning the reactor cavity. The assemblies are transported in the cask through an air lock in the secondary containment vessel to a fuel-storage facility.

HWR-U. The reactor is shut down for refueling about every four months and one-third of the 516 fuel assemblies replaced. Fuel assemblies remain in the same location throughout their life in the core. The fuel elements are removed by a refueling machine that operates inside a shielded crane bay over the reactor and fuel transfer pit. The control rod drives and supports must be removed first to permit removal of a shielding slab for access to the pressure-tube nozzles. During refueling, the machine must remove each pressure-tube closure remotely, change the fuel assembly, and replace the closure. After removal, a fuel assembly is carried by the refueling machine to a transfer pit filled with water. Another fuel-handling machine operating over the transfer pit loads the irradiated fuel assemblies into a mechanism for transfer through an underwater lock to a storage facility outside the secondary containment vessel. Very few details were available pertaining to the design and operation of this equipment.

HWR-Th. Fuel handling is performed while the reactor is in operation. The three concentric fuel tubes contained in each pressure tube are separated axially into two or, possibly, three assemblies having a combined length of 15 ft. To avoid interference with the control rod drives located above the core, the fuel-handling machine operates in a space below the reactor. The machine attaches to the pressure-tube nozzle, removes the closure, changes the fuel, and then replaces the closure; this operation is carried out at the system pressure and temperature. Fuel is stored temporarily in the refueling machine while a turntable lines the machine up with stationary tracks for movement from under the core. Assemblies are next transferred from the refueling machine through locks to a conveyer and are then placed in a transfer canal for movement to storage facilities outside the secondary containment vessel.

This equipment must be extremely reliable to permit continuous operation of the plant, since the refueling machine performs a shim function and thus acts as a reactor control device. The reactor cannot operate over extended periods while the refueling machine is out of service. Operation of the fuel-handling equipment will be difficult since it must be carried out remotely in a high radiation field. It is possible that the machinery can malfunction in any phase of the refueling operation; direct maintenance will be impossible when irradiated fuel is inside, since it appears as though the machine is not provided with a biological shield.

A detailed study of the proposed design concept based on actual detailed design and operating procedures would be required in order to establish that a particular concept can operate with the necessary reliability. The successful operation of the Canadian NPD refueling machine and apparently satisfactory development of equipment for on-line refueling of the 200-Mw Douglas Point Plant suggest, however, that such equipment can be considered feasible.

HTGR. The reactor is shut down about every six months for refueling, and 1/12, or 458, of the 5489 fuel elements is replaced. Fuel elements remain in the same position throughout their life in the core, and those replaced are in clusters that are fairly uniformly distributed over the core. The clusters are selected to correspond to refueling stations, and eleven stations are refueled at each shutdown. All the fuel elements serviced by a refueling station are replaced at the same time. The refueling operations are carried out through 127 nozzles, all but 6 of which also serve as control rod nozzles. The fuel handling is accomplished by means of a refueling machine that operates on a bridge spanning the upper face of the reactor vessel. Primary system pressure is reduced to essentially atmospheric for the refueling operation. The refueling machine first removes the control rods and control rod drives in order to gain access to the fuel and then removes fuel assemblies and transports them through an air lock to a fuel storage facility. The refueling machine has a storage capacity of two control rods and five fuel elements or seven fuel elements. Thus about nine round trips are required to change the fuel in one cluster consisting of two control rods and 47 fuel elements. The fuel storage facilities are designed to accommodate the normal flow of fuel assemblies to and from the reactor, and no on-site provisions have been made for temporary storage of an entire core inventory. It appears as though it would be necessary to provide temporary storage if an irradiated core had to be removed for in-core maintenance operations.

The extremely close clearance of 0.001 in. specified between the fuel element and standoff pin does not appear to be realistic for remote installation of the fuel. The degree of precision required to assure proper alignment will impose combined dimensional tolerances extremely difficult to achieve, and a small amount of deflection or distortion of the pins may prevent removal or installation of a fuel element. Remote replacement of pins to required tolerances may also present difficulties.

Some problems of core disassembly are presented by the use of multiple spacers along the length of the fuel element. The close-fitting spacers must be capable of passing each other to permit removal of the first few elements by direct vertical lift. The dimensions and clearances required for the element are:

Fuel assembly pitch, in.	4.700
Top spacer outside diameter, in.	4.700
Nominal top spacer clearance, in.	0.000
Intermediate spacer outside diameter, in.	4.690
Nominal intermediate spacer clearance, in.	0.010
Nominal lower spacer clearance, in.	0.010

These clearances are quite small considering the overall length of the fuel assembly. It is expected that the accumulation of reasonably achievable fabrication tolerances of straightness, concentricity, and diameter variation would be two to three times the nominal clearance specified for an assembly of this size.

This problem is compounded by the bowing and distortion of the elements after operation, but it could be alleviated to some extent if the

spacers were not located on equal centers as in the proposed spacer arrangement. If this were the case it would not be necessary to pass two tight-fitting locations simultaneously. This would permit using the increased clearance at one end to wiggle the other end through the small clearance between spacers. In either case the presence of the two lower spacers will probably cause difficulty in removing the fuel. Should the interference occur, one or more of the adjacent fuel elements might be pulled out of the core with the one being handled. The element would become dislodged after it was lifted high enough to shift and relieve the interference, and it would then fall back into the core. This could result in damage to the element or require special equipment to retrieve the element. Additional design studies may provide a satisfactory solution to this problem.

SGR. The SGR is shut down every 8 months for refueling. One-sixth, or 99, of the 594 fuel assemblies is changed and approximately 50 assemblies are relocated at each refueling shutdown. A refueling machine similar to that in the Hallam plant is used to replace the fuel assemblies. The refueling machine is mounted on a gantry crane that operates over the reactor face and fuel storage facility. Fuel assemblies are replaced through individual penetrations in the upper reactor shield. The machine transports the spent fuel to a storage facility and picks up new fuel at a loading station. A round trip must be made for each fuel assembly replaced. Spent fuel is stored in rotary fuel magazines located in sodium-filled pits that are contained and shielded similarly to the reactor.

Plant Availability

The overall plant availability, refueling, and maintenance schedule will to a large extent determine how well a particular type of nuclear plant can be integrated into the power system in which it is to be used. Availability requirements will vary from system to system. An attempt is made here to estimate the overall plant availability for each of the reactor types under consideration. The availability factor is intended to indicate the composite effects of

1. Reactor refueling down time and frequency,
2. Reactor periodic maintenance and inspection down time,
3. Steam plant periodic maintenance and inspection down time,
4. Steam plant equipment reliability and repair down time.

The plant availability is defined here as the product of four separate factors covering the categories indicated above. These factors are determined on the basis of the proposed fuel management scheme, estimated refueling time, and nominal maintenance requirements. The following rules and assumptions were applied to get an overall indication of plant availability. The assumptions are somewhat arbitrary but are intended to be realistic.

Refueling Factor. The refueling factor is based on the specified fuel management scheme and the estimated refueling time as previously defined in the section pertaining to fuel handling.

Nuclear Plant Maintenance Factor. All plants are assumed to be shut down for at least one week out of every year for periodic maintenance and

inspection of reactor plant equipment. The same requirement is imposed on the steam plant equipment, but no additional time is needed, since both can be carried out concurrently. Plants that must shut down seven days or more for refueling every year are not penalized for this requirement, since the refueling down time is adequate for this purpose.

Steam Plant Scheduled Maintenance Factor. It is assumed that the plant must be shut down at least 30 days out of every two years operating time for major overhaul of the turbine-generator unit. The steam plant is only charged for the additional time required over and above that required for scheduled nuclear plant shutdown.

Steam Plant Reliability Factor. A factor of 0.977 is applied to all plants to cover forced outages attributed to steam plant and power transmission equipment.

Under the assumptions used, the maximum possible plant availability factor would be 0.931 if the reactor were not penalized for refueling or forced outages due to nuclear plant equipment. As a point of comparison, a 1955 ASME article[5] cites an average availability of 0.918 for the most efficient units of 12 different power stations. Actually the availability factors of conventional plants have tended to decrease with the increasing size of units and increased temperatures, but no attempt was made here to distinguish differences in steam conditions or turbine types. A comparison of the availability factors estimated for the various power plants is tabulated above in Table 5.3.

In a sense the availability factor is a measure of how well the fuel management scheme, capital investment, maintenance cost, and power output are balanced in the overall plant design. The plant availability can be changed significantly by altering the fuel management scheme or investing additional capital in refueling equipment or a combination of the two. The implication of a low availability factor is that (1) the fuel management scheme and refueling interval should be more closely coordinated with the requirements of the rest of the system, and (2) a modification of the fuel-handling approach and equipment may be necessary.

The plant availability in conjunction with the demand made on the plant will determine the load factor. The availability is a characteristic of the individual plant, and the demand is characteristic of the system in which the plant is used and depends also on how well the particular plant can be integrated into the system. This would vary with the utility system as well as the reactor plant. In order to compare load factors it is assumed that the plants are base loaded and must be capable of an average availability of 0.90 over the life of the plant in order to qualify for an 80% load factor.

The plant availability factors indicated in Table 5.3 assume that the reactor plant equipment is 100% reliable and no forced outages are attributed to this equipment. Based on this additional assumption, all the reactor systems under consideration could qualify for an 80% load factor.

The availability factors indicated represent maximum values and, in reality, would be expected to be somewhat less than those indicated due to forced outages of the power plant attributed to failures or uncertainties in the operation of the nuclear plant equipment. The PWR, SSCR, and HWR-Th plants have some margin over the minimum 0.90 factor to cover a nomimal amount of difficulty with nuclear plant equipment. However, the HWR-U, HTGR, and SGR appear to be marginal in their ability to carry an

80% load factor, since minor difficulties in the nuclear plant would drop their average availability below the required 0.90.

Control Rod Systems

It is difficult to comment on the technical feasibility of the control rod systems from the information given. Assumptions used for relative cost comparisons of these systems are discussed in Chapter 6.

Heat Transfer Systems

Design specifications for heat transfer system components are listed in Table 5.4. Although all reactor plants do not have the same number of coolant loops, there is substantial agreement in the water-cooler reactors on the selection of pumps and steam generators with similar total mass flow rates, temperatures, and heat transfer areas (with the exception of the lower heat transfer area of the SSCR noted later).

The primary coolant pump design used for the water-cooled reactors is a vertically mounted, centrifugal pump using an injection-type low-leakage mechanical seal. This type of pump has good reliability and is considerably less expensive than the canned-motor pump used extensively in the past. Pumps of the capacity and developed head shown in Table 5.4 are currently available for system pressures up to about 1000 psi. Availability of pumps for the better than 2000-psi reactor pressures depends upon development of satisfactory seals. Multistage seals for pressures in excess of 2000 psi have been developed for shafts somewhat smaller than required for these pumps. It is thought that larger seals are attainable, and the amount of development work required to demonstrate the suitability of using them for the water-cooled reactors should be relatively small. The use of these pumps is therefore assumed for normalization of the PWR, SSCR, and HWR primary coolant circulating systems.

The SGR also uses a vertically mounted centrifugal pump. This pump has a free surface of sodium above the impeller and lower bearing. The volume above the sodium is blanketed with helium, and a double mechanical seal is provided at the upper cold end of the pump for atmospheric separation. The same pump is used in both the primary and secondary coolant loops. Although the pump is approximately four times larger than sodium reactor pumps used in the Hallam plant, no difficulty is foreseen in scaling up.

The HTGR cooling loops use vertically mounted steam-turbine-driven axial-flow gas circulators installed in the reactor vessel. Both the turbine and the compressor are overhung from the circulator center housing, with the compressor end in the primary helium containment vessel and the steam turbine end in the secondary containment vessel. The central housing contains the thrust and journal bearings and the oil and gas seals. Although the compressor and turbine aerodynamic designs are based on acceptable industrial standards, the circulator is unique in its method of installation and in the supporting and sealing of the common turbine-compressor shaft. A prototype circulator of this type is being developed currently to demonstrate its feasibility.

Table 5.4. Reactor Heat Transfer Systems[a]

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Steam generators						
Number of steam generators	7	5	6	6	6	5 circuits, 54 modules per circuit
Type of construction	Vertical U-tube, integral steam drum	Vertical U-tube	Vertical U-tube, integral steam drum	Vertical U-tube, integral steam drum	Vertical bundle, no shell	Vertical shell and tube, module type
Coolant	H_2O	$D_2O + H_2O$	D_2O	D_2O	Helium	Sodium
Reactor coolant flow rate, lb/hr	160×10^6 total, 22.86×10^6 each			150×10^6 total, 25.0×10^6 each	8.32×10^6 total, 1.39×10^6 each	66×10^6 total
Reactor coolant inlet temperature, °F	598	594	579	579	1470	1060
Reactor coolant outlet temperature, °F	546	535	507	507	720	660
Steam temperature, °F	503		471	471	1050	1005
Feedwater temperature, °F	442		387	387	547	512
Reactor coolant pressure, psia	2500 design, 2050 operating	2500 design, 2200 operating	2000 design, 1700 operating	2000 design, 1700 operating	450 operating	48 operating
Steam pressure, psia	1000 design, 700 operating		510 operating	510 operating	3800 design	3615 operating
Heat transfer rate, Btu/hr	11.0×10^9 total, 1.57×10^9 each		12.4×10^9 total, 2.1×10^9 each	12.4×10^9 total, 2.1×10^9 each	7.88×10^9 total	8.05×10^9 total
Heat transfer area, ft^2	229,000 total, 32,726 each	145,000 (229,000) total, 29,000 (45,800) each	228,000 total, 38,000 each	228,000 total, 38,000 each	204,800 total	167,800 total
Number of tubes	Not given		60,900 total, 10,150 each	60,900 total, 10,150 each	10,464 total	
Size of tubes, in.	Not given		1/2	1/2	(b)	
Tube material	Inconel	Inconel	Inconel	Inconel	(c)	(d)
Shell material	Carbon steel	Carbon steel	Carbon steel	Carbon steel	None	Same as tubes
Primary piping	Stainless steel-clad carbon steel	Type 316 stainless steel	Carbon steel	Carbon steel	Not applicable	Type 304 stainless steel
Pressurizers						
Number of pressurizers	1		1	1	None	None required, uses surge tank
Pressure, psig	2035		1700	1700		
Temperature, °F	639					
Inside diameter	10 ft 4 in.		10 ft	10 ft		
Shell thickness, in.	8 1/2		7 1/2	7 1/2		
Length, ft	50		26	26		
Material	Stainless steel-clad carbon steel	Stainless steel-clad carbon steel	Carbon steel	Carbon steel		

[a] All values in parentheses are for normalized or assumed conditions.

[b] 3/4 in economizer-evaporator; 5/8 in superheater; 1 in reheater.

[c] 1 1/4 Cr–1/2 Mo steel in economizer-evaporator; 300-series stainless steel in superheater and reheater.

[d] 2 1/4 Cr–1 Mo in low-temperature modules; type 321 stainless steel in high-temperature modules; type 304 stainless steel in reheater modules.

Table 5.4 (continued)

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Recirculating pump and motors						
Number of pumps	7	5	6	6	6	5
Pump type (canned motor)	(Mechanical seal, centrifugal)	Mechanical seal, centrifugal	Mechanical seal, centrifugal	Mechanical seal, centrifugal	Steam turbine, axial flow	Mechanical seal, centrifugal
Capacity	61,000 gpm per pump	76,200 gpm per pump	53,500 gpm per pump	53,500 gpm per pump	8.32×10^6 lb/hr total flow	33,000 gpm per pump
Head, ft	280		556	556		190
Coolant temperature, °F	546		507	507	720	1150
Shell pressure, psia	2500 design, 2150 operating		2000	2000	450	
Pump material	300-series stainless steel	Stainless steel				Type 304 stainless steel[e]
Driver size, hp	4800		10,000	10,000		2250
Type of driver	Motor	Motor	Motor	Motor	Steam turbine	Motor
Intermediate heat exchanger						
Number of heat exchangers						2 for each of 5 circuits
Type of construction						Modular shell and tube
Primary coolant flow rate, lb/hr						66×10^6 total
Primary coolant inlet temperature, °F						1150
Primary coolant outlet temperature, °F						750
Secondary coolant outlet temperature, °F						1060
Secondary coolant inlet temperature, °F						660
Heat transfer rate, Btu/hr						8.75×10^9 total, 0.875×10^9 each
Heat transfer area, ft^2						98,900 total
Number of tubes						28,800 total
Size of tubes, in.						5/8
Tube material						Type 304 stainless steel
Shell material						Type 304 stainless steel

[e]Not certain.

All the water-cooled reactors use steam generators of the vertical U-tube type with Inconel tubes and carbon steel shells. Although Inconel tubing is more expensive than the stainless steel tubing frequently used in this service, part of the higher cost is offset by the higher allowable stress for Inconel, which permits thinner tubes. A principal reason for the use of Inconel is that it can resist high concentrations of chloride ion without stress corrosion cracking and thus assure high reliability of the steam generator.

The PWR and heavy-water reactor steam generators have an overall heat transfer coefficient of approximately 750 $Btu/hr \cdot ft^2 \cdot °F$, whereas the SSCR has a much higher value of 1305. To maintain a comparable design basis, steam generators having a total heat transfer area equal to the PWR are imposed upon the SSCR with appropriate cost adjustment.

The imposed increase in SSCR steam-generator size increases the inventory of primary coolant by an estimated 50,000 lb of water. The containment vessel size has been increased to hold this added coolant at the original equilibrium pressure.

Both the HTGR and SGR supercritical steam generators are of once-through design, with low-temperature, high-temperature, and reheat sections. The HTGR generators are incorporated into the prestressed-concrete reactor vessel (PCRS) eliminating the conventional shell and primary piping. The SGR generators consist of small modules built up of once-through tubes and standard pipe shells connected together externally that allow simplified fabrication, installation, and maintenance. Conventional shell-and-tube type heat exchangers are used for the SGR intermediate coolant system.

Primary coolant purification systems consisting of demineralizers, oxygen, and pH control are used with the water-cooled reactor systems. The HTGR requires an extensive system for helium purification in which a purge flow from the fuel elements is fed to a bypass loop for removal of entrained water and delay of xenon and krypton. Most of this stream is returned to the coolant system, but part is purified additionally to remove CO, CO_2, N, Xe, Kr, and H. The HTGR fuel-element purge uses a smaller purification flow than some side-stream purification concepts, but it is limited in the amount of water inleakage that can be handled. However, the allowable water inleakage is probably limited by the tolerable damage to the graphite rather than the size of the purification system. The SGR has an extensive system for storing and purifying the sodium coolant that incorporates hot and cold traps, transfer and venting systems, and storage tanks.

Material specifications for the heat transfer systems are, in general, conventional for the coolant types and service conditions. The heavy-water reactor utilization of carbon steel for primary system piping is somewhat unconventional. However, there are experimental data and precedent to indicate that carbon steel can be a satisfactory material, provided the proper pH is maintained, oxygen in the coolant is restricted to a low level, and the coolant is not stagnant. When the additional cost of coolant quality control is combined with the cost involved in attaining a carbon steel system having material and fabrication quality control at a level comparable with stainless steel systems, much of the economic advantage vanishes without achieving the integrity of stainless steel systems.

Plant auxiliary systems naturally vary with the type of reactor. Essential systems for each of the reactor plants seem to have been considered. No critical review of auxiliary components is attempted due to the lack of detailed information.

Heavy-Water Losses

Operating data from the Plutonium Recycle Test Reactor at Hanford have shown that the heavy-water leakage rate can be controlled to 16 lb per day for a pressure-tube reactor with 85 tubes and that about 95% of the leakage is from the high-pressure system. Further improvements are expected by the installation of improved leak-detection equipment in the off-gas system. Because of the flexibility required of the PRTR pressure tubes, each tube has three Parker fittings and three gaskets through which D_2O may escape. Hence it is expected that the loss rate of a power reactor, which would have welded fittings rather than Parker fittings, would be lower. If it is assumed that the loss rate could be cut to 12 lb/day for 85 pressure tubes, the annual loss rate per tube would be 51.5 lb. For the HWR-U with 516 tubes and a heavy-water inventory of 1.355×10^6 lb, the loss rate would then be 1.96%; for the HWR-Th with 688 tubes and a heavy-water inventory of 1.644×10^6 lb, the rate might be 2.16% per year.

On the basis of the above, an annual loss rate of 2.0% of the total heavy-water inventory appears to be reasonable. However, the on-line fuel-charging machine for the HWR-Th must not add significantly to the loss rate.

Instrumentation and Controls

All the reactor plants have the standard instrumentation for monitoring flow, temperature, and radioactivity of the coolant; nuclear instrumentation for start up, reactor period, and high-level flux monitoring; and primary cooling system instrumentation for measuring and controlling coolant flow, pressure, and level. All reactors, except the SGR, make use of in-core instrumentation to obtain information about temperature and flux distribution, flow, or radioactivity of the coolant. This is applied most extensively to the PWR with temperature measurement of 100 fuel assemblies and 67 movable ion chambers and to the heavy-water reactors, where flow, temperature, and radioactivity are measured at each fuel position (688 for the HWR-Th and 516 for the HWR-U). Even though the extensive in-core instrumentation adds a cost burden to some of the reactors, it is believed that the relative complexities of the systems are in line with the nuclear, control, and thermodynamic characteristics of the particular reactors.

Containment Structures

Design data for the containment structures are presented in Table 5.5. Containment for the water-cooled reactors is based on loss of all primary coolant, which immediately vaporizes and combines with the release of all fission products from a core meltdown. Reference design information on

Table 5.5. Reactor Containment Structures[a]

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Design basis	Contain primary coolant and complete core melt-down	Contain primary coolant and complete core melt-down	Contain primary coolant and secondary coolant in one steam generator	Contain primary coolant and secondary coolant in one steam generator	Contain release of 9400 lb of helium	Reactor, fuel-handling, and primary systems in steel-lined, concrete-shielded cells filled with nitrogen atmosphere
Type of construction	Steel-lined concrete cylinder with hemispherical head	Steel-lined concrete cylinder with hemispherical head	Spherical, steel (cylindrical concrete assumed)	Spherical, steel (cylindrical concrete assumed)	Steel-lined concrete cylinder with hemispherical head	Concrete 5 ft thick
Inside diameter, ft	162	125 (130)	260 (162)	250 (162)	135	
Total height, ft	184 (216)[b]	165 (191)	260 (201)	250 (201)	125	
Wall thickness	5 ft 6 in.	4 ft 6 in.	3/4 in. (5 ft 6 in.)	1 1/4 in. (5 ft 6 in.)	(2 ft 6 in.)	
Head thickness	5 ft	3 ft 6 in.	3/4 in. (5 ft)	1 1/4 in. (5 ft)	(2 ft 6 in.)	
Total volume, net, ft^3	3,500,000	1,588,000 (2,010,000)[c]	8,290,000 (3,231,000)[c]	7,362,000 (3,231,000)[c]	1,475,000[c]	
Maximum internal pressure, psig	40	39	17.5 (36)	25 (40)	(21.5)	
Pressure suppression	None	Spray system	Deluge system	Deluge system	None	None
Cleanup system	Containment volume filtered before venting to atmosphere	Containment volume passes through filters and adsorbers before venting to atmosphere	Containment volume passes through filters and adsorbers before venting to atmosphere	Containment volume passes through filters and adsorbers before venting to atmosphere		

[a]All values in parentheses are for normalized or assumed conditions.

[b]Height changed to obtain reported volume.

[c]Estimated net volume based on 90% of gross volume.

the heavy-water reactors also includes the loss of water in the secondary side of one steam generator.

The large variation in reported size of containment vessels is due primarily to the plant layouts reported. Generally, the most economical containment is attained with a small vessel operating at the highest pressure it can stand without getting into fabrication problems due to the heavy wall thickness (such as the stress-relief requirement for carbon steel beyond certain thicknesses).

The PWR and SSCR have steel-lined concrete vessels, while the heavy-water reactors have carbon steel vessels. Both have been used satisfactorily, with concrete vessels usually having the advantage of lower cost.

To put the heavy water reactors on a comparable basis with the other water-cooled reactors, it was assumed that the primary reactor system equipment could be arranged in a smaller vessel (162 ft in diameter) having cylindrical sides and using a higher equalization pressure. The material was changed to reinforced concrete to take advantage of its lower cost. Unless space requirements associated with HWR fuel handling prohibit reducing the area within the containment structure to the assumed size, the assumption used for this normalization appears to be valid.

All vessels except the PWR use spray systems for condensing the vaporized coolant quickly after a release and reducing the pressure and hence the leakage through the containment to atmosphere. All containment systems provide for filtering and adsorbing radioactive gases from the contained volume prior to venting through the stack to the atmosphere.

It is claimed for the HTGR design with a prestressed-concrete vessel that a catastrophic failure is incredible, since the vessel would not fragment as would a steel vessel. On that basis, even the worst failure would be of such size that the leak could be handled by a leak-collector system, and conventional containment would not be required. Even though sizeable savings could result from this approach, the proponent acknowledges that there is not sufficient experience to adequately establish the safety capabilities. Therefore, an alternate design is presented for a conventional steel-lined concrete containment vessel to hold the release of all helium in the system. This alternative is assumed for the present design and cost normalization. No consideration was given a loss of the secondary coolant (water) either to the building or to the reactor.

The low operating pressure and temperature of the SGR limit the stored energy in the primary system, so there is no concern about vaporizing the coolant. Any evolved gas from a fuel element is picked up in the helium blanket above the sodium volume and disposed of through the radioactive vent system. An advantage of the SGR is the use of a conventional industrial building rather than a containment structure.

Turbine-Generator Systems

Design and performance features of the turbine-generator systems for the PWR, HTGR, and SGR plants are compared in Table 5.6. Since no equipment requirements were furnished for the SSCR and HWR concepts, it was assumed in this study that the requirements were the same as those for the PWR, except for minor variations to allow for different throttle steam pressures and feedwater return temperatures. All plants use an

Table 5.6. Design and Performance Features of Turbine-Generator Systems[a]

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Turbine-generator						
Throttle pressure, psia	650	600	495	495	3515	3515
Temperature, °F	495	490	466	466	1050	1000
Enthalpy, Btu/lb	1200.5	1202	1204.5	1204.5	1461.1	1424.0
Flow, 10^6 lb/hr	13.146	(b)	(b)	(b)	6.886	7.237
Reheat pressure, psia	108	(b)	(b)	(c)	718.5	691.3
Temperature, °F	466.5	(b)	(b)	(c)	1050	1000
Enthalpy, Btu/lb	1216.2	(b)	(b)	(c)	1540.8	1514.1
Flow, 10^6 lb/hr	9.544	(b)	(b)	(c)	6.192	5.220
Condenser pressure, in. Hg abs.	1.5	1.5	1.5	1.5	1.0 (1.5)	1.5
Flow, 10^6 lb/hr	7.942		8.940	8.940	4.314	4.516
Enthalpy, Btu/lb	981.5				985.6 (1002)	993.1
Duty, 10^9 Btu/hr	7.795				4.252 (4.322)	4.485
Turbine arrangement	TC6F—50-in. LSB	(TC6F—50-in. LSB)	(TC6F—43-in. LSB)	(TC6F—43-in. LSB)	CC4F—52-in. LSB (43-in. LSB)	CC4F—43-in. LSB
Net station output, Mw	1002.4	1000	1003	1010	1039 (1008)	1019
Net station heat rate, Btu/kwhr	10,953	10,939	12,735	13,077	7451 (76.87)	7824
Net station efficiency	31.1	31.2	26.8	26.1	45.8 (44.4)	43.6
Reactor thermal output, Mw	3223	3205	3750	3870	2270	2336
Station electrical auxiliary power, Mw(e)	55.6	(51)	80	70	21	31.5
Generator rating, total, kva	1,200,000	(1,200,000)	(1,200,000)	(1,200,000)	1,254,000	1,280,000
Terminal voltage, kv	26	(26)	(26)	(26)	24	24
Power factor	0.90	(0.90)	(0.90)	(0.90)	(d)	0.85
Hydrogen pressure, psi	60	(60)	(60)	(60)	(d)	4.5
Feedwater temperature, final, °F	442.7	(435)	387	387	537.6	511.8
Enthalpy, Btu/lb	422.2	413.3	361	361	530.7	501.0
Shaft-driven auxiliaries	None	(None)	(None)	(None)		
Exhaust-steam-powered auxiliaries	None	(None)	(None)	(None)	Blowers[e]	Boiler feed[f] pumps

[a]Value in parentheses is for normalized conditions or assumed.

[b]No heat balance.

[c]No reheat assumed.

[d]Not applicable.

[e]Blower turbine exhausts to main turbine reheater.

[f]Pump turbine exhausts to feedwater heaters.

Table 5.6 (continued)

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Condensing system						
Circulating-water pumps	6	(6)	(6)	(6)	4	4
Flow rate, each, gpm	110,000	(110,000)	(129,000)	(129,000)	176,000 (107,300)	107,250 (112,300)
Head, ft	28	(28)	(28)	(28)	(28)	30 (28)
Drive rating, hp	900	(900)	(900)	(900)	(d)	900
Condenser type	Single pass	Single pass	Single pass	Single pass	Single pass	Single pass
Units	3	(3)	(3)	(3)	2	2
Tubes, BWG	18	(18)	(18)	(18)	(18)	18
Material	Admiralty	(Admiralty)	(Admiralty)	(Admiralty)	(Admiralty)	Admiralty
Size, in.	1	(1)	(1)	(1)	(7/8)	7/8
Length, ft	50	(50)	(50)	(50)	(40)	38 (40)
Design heat transfer surface, ft^2	579,000	(579,000)	(660,900)	(660,900)	530,000 (312,000)	334,000 (336,200)
Design cooling-water temperature, °F	57	(57)	(57)	(57)	53 (57)	57
Design saturation temperature, °F	91.7	91.7	91.7	91.7	79.1 (91.7)	91.7
Cooling-water flow, total, gpm	660,000	(660,000)	(772,000)	(772,000)	704,000 (429,200)	429,000 (449,200)
Feedwater system						
Feedwater demineralizing	None	(None)	(None)	(None)	Full flow	Full flow
Deaeration	No	(No)	(No)	(No)	Yes	Yes
Feedwater heaters, total number	18	(18)	(18)	(18)	14	16
Number of banks	3	(3)	(3)	(3)	2	2
Drains	Cascade	(Cascade)	(Cascade)	(Cascade)	Cascade	Cascade
Boiler feed pumps, number	3	(3)	(3)	(3)	3	3
Capacity	1/3	(1/3)	(1/3)	(1/3)	One at 75%, two at 12 1/2% each	50% each
Flow, gpm	11,400				One at 11,500, two at at 1950 each	8000 each
Feedwater temperature, °F[g]	334	(334)	(334)	(334)	340	325
Total dynamic head, ft	1700					11,400
Driver	5000-hp motors	5000-hp motors	5000-hp motors	5000-hp motors	Main turbine	Turbine
Emergency boiler feed pumps, number	1	(1)	(1)	(1)	1	(1)
Capacity					12 1/2%	
Drive	Steam driver	Steam driver	Steam driver	Steam driver	Motor, steam turbine powered	(Motor, steam turbine powered)

[g]Not final.

indirect (and nonradioactive) steam cycle. Construction is conventional and directly comparable with existing fossil-fuel and nuclear plant turbine-generator systems.

The PWR system utilizes a saturated steam cycle with external moisture removal and live steam reheat ahead of the low-pressure turbines. The thermal efficiency of the PWR and the other water-cooled reactor plants is limited to about 31% because superheating of steam delivered to the high-pressure turbine by reactor power is not provided for in these concepts.

The heat transferred versus steam temperature for each of the water-cooled reactor concepts is shown in Fig. 5.1. Based on pinch-point temperature differences at the economizer outlets, heating surface requirements are about the same for all the water-cooled plants. State points for the turbine expansions indicate that all plants follow the conventional practice of limiting moisture in the exhaust steam to about 12%.

The PWR design proposes a single tandem-compound turbine-generator for the full 1000-Mw(e) net output. Maximum efficiency is obtained under present material technology by the use of 50-in. last-stage buckets. It is to be noted that this turbine-generator unit is larger than any tandem compound units on order to date. Should there be any problem in development of this unit, replacement by two tandem-compound units of 500-Mw capacity each could be made with a penalty on capital cost.

The HTGR and SGR plants both employ supercritical steam cycles with superheat. The operating temperatures of 1050°F for the HTGR and 1000°F

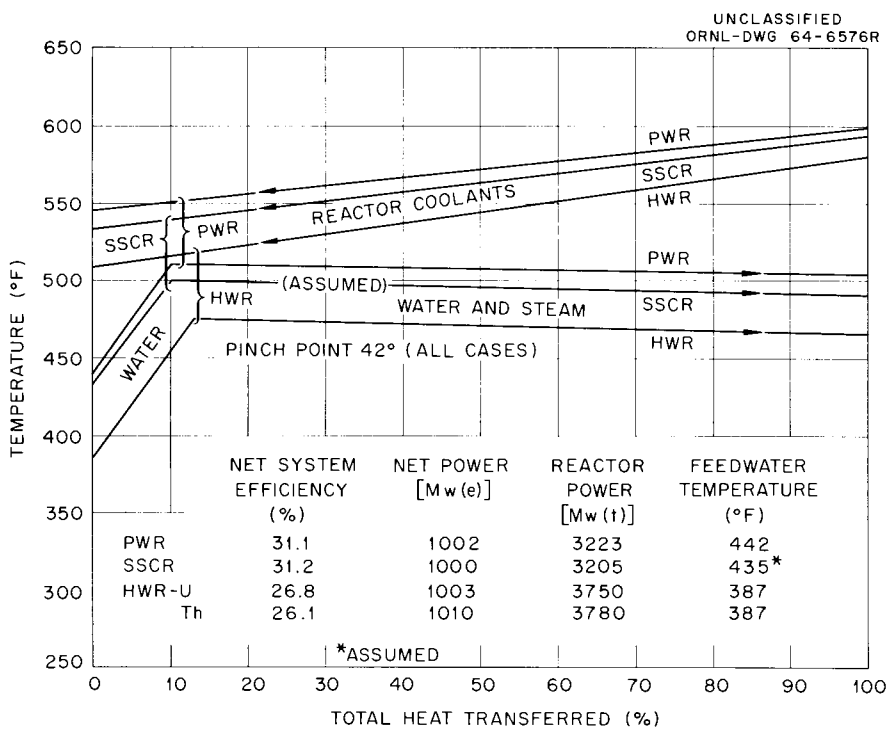

Fig. 5.1. Heat Transferred as a Function of Temperature in Steam Generators of PWR-Type Systems.

for the SGR should pose no unusual problems, since reliability of materials in this temperature range is well established. Thermal efficiencies of about 45% can be realized in these plants.

The HTGR and SGR designs propose conventional cross-compound turbine-generators with provisions made in the HTGR design for supplying steam to the turbine driving the primary coolant blowers. However, the HTGR design specifies 52-in. last-stage bucket blades, as compared with 43-in. blades with a 1-in.-Hg back pressure in the SGR design. The 52-in. design provides improved efficiency over the 43-in. design, but its reliability is not established in practice.

The HTGR design specifies a 1.0-in.-Hg back pressure on the turbine instead of the 1.5-in.-Hg pressure given in the ground rules for this evaluation study. The effect on efficiency and equipment sizes of changing to 1.5-in.-Hg pressure is accounted for in the cost normalizations in the next section of this report. Essentially, the effect of changing to 1.5-in. Hg reduces the net electrical output by 21 Mw and reduces the condenser size approximately 25%.

In addition, the advantage in efficiency gain of using 52-in. instead of 43-in. last-stage buckets is reduced by about one-half at 1 1/2-in.-Hg as compared with 1-in.-Hg back pressure. This gain in efficiency is more than offset by the higher capital cost; therefore, for the purpose of this study it was assumed that the 43-in. design would be used for both plants. This change in design results in a further reduction in electrical generation of 10 Mw. The net efficiency of the HTGR plant would be reduced from 45.8 to 44.4%, due principally to the combined change of turbine-back-pressure and turbine design.

A much more reliable emergency feedwater supply system is required for the HTGR plant than for the SGR to accommodate reactor shutdown cooling. To meet such a need, the HTGR design provides a self-contained turbine-generator set operated off steam generated by shutdown heat to drive the feedwater pumps.

In comparing auxiliary and spare equipment provisions in the reference turbine-generator system designs, it is found that each follows normal industrial practice in most respects. One exception is that the PWR design, which is assumed for all the water-cooled plants, does not include provision for a spare exciter, although a "spare-in-kind" is furnished by using two half-size exciters.

Comparisons of important considerations, such as pipeline velocities, which affect system optimizations, could not be made in this study because only the PWR system is adequately described to permit it.

In summary, conventional turbine-generator systems can be used in each of the reactor concepts. There are relatively minor variations in equipment requirements among the water-cooled reactor plants due to different steam and feedwater return conditions that are set by the reactor coolant temperatures. These variations can be readily taken into account in capital cost normalization. The supercritical steam cycles of the HTGR and SGR also have conventional equipment. For normalization of costs it is assumed that both will use the same type of turbine-generators, with a resultant change in HTGR net efficiency of from 45.8 to 44.4%.

Electrical Facilities

Accessory electrical facilities for the plant services are not defined for all cases; therefore, designs for the two groups of reactor plants are assumed to be based on those given for the PWR and HTGR. It is expected that systems that are identical except for capacity will be suitable for all cases. Assumptions used in the cost normalization are detailed in Chapter 6.

Site and Buildings

Site adjustments were required in all designs to conform to the AEC reference site. The modifications affecting normalization are outlined in Chapter 6. In general, the common site, Middletown, consists of 1200 acres of grass-covered level terrain adjacent to a river with adequate flow for cooling-water requirements. Ground elevation is about 40 ft above low water and 20 ft above high water. A limestone foundation exists 8 ft below grade.

Ground improvements are standardized on the PWR concept, which appears to be adequate for the other plants.

Building requirements, with the exception of the reactor containment, which is described separately, are considered under two groups. The first is the PWR plant design, in which building layout is given in considerable detail. The housing functions are adaptable for the other water-cooled plants, which are not accounted for in any degree by the designers. For both high-temperature reactors, specific reactor auxiliary service building requirements are reported that are retained as given. The remaining structures, which house the turbine-generator, offices, shops, and control rooms, are normalized on the basis of the PWR concept, since it is suitable for the housing functions of the HTGR and SGR plants.

Summary and Conclusions

The design features of most of the systems of the water-cooled reactor plants are amenable to direct comparison and normalization. Achievement of performance requirements for the major components of these systems appears to be within the scope of established technology or reasonable extension of existing technology. The feasibility of each of these concepts depends largely on the successful solution of problems associated with (1) the reactor vessels, which are different for each plant, and (2) the refueling machines in the case of the heavy-water reactors.

The PWR pressure vessel presently specified is smaller and has thinner walls than the vessel earlier thought to be required for the 1000-Mw(e) plant. The development program to establish fabrication methods for the presently required 11 1/8-in.-wall vessel should be considerably less extensive than that described by Combustion Engineering Co. for the original 13 7/8-in.-wall vessel. Some development work is required on PWR core internal structure to demonstrate achievement of the flow distribution required of the design.

The cylindrical SSCR vessel resembles the PWR vessel, but it is smaller and should be much easier to build.

The HWR pressure-tube vessel concept requires successful demonstration of satisfactory metallurgical properties of irradiated Zr-2.5% Nb tubes. Experimental information on corrosion and creep appear favorable but require an excessive amount of extrapolation to predict a 30-year behavior. It may be necessary to increase the thickness of the HWR-U pressure tube above the value presently specified to allow for corrosion. Reliable mechanical closures and seals must be developed for the pressure tube to minimize D_2O leakage and provide for remote refueling.

The HWR-Th concept depends on the development of a machine for remote on-line refueling. High reliability of performance under severe conditions is required of this machine to avoid shutdowns that would adversely affect plant availability.

The HTGR and SGR plants have supercritical steam cycles that can be compared with each other and with existing fossil-fuel stations. Other than for the steam cycles, systems and equipment of these plants are not comparable with other concepts of this study.

The HTGR plant design feasibility depends largely on the successful demonstration of the prestressed-concrete reactor structure (PCRS) that houses the pressurized primary system, including the reactor, helium-circulating blowers, and steam generators. The PCRS test model program currently under way at General Atomic will provide information on this design. British experience with the Oldbury prestressed-concrete reactor vessel model is also applicable to the HTGR concept, as is French experience with prestressed-concrete vessels. The Oldbury reactor station, which is scheduled to go into operation in 1966, will provide information on the features of this design before the HTGR would be built. The HTGR design also depends on the development of satisfactory seals and bearings for the vertical shafts of the turbine-driven axial flow compressors.

The General Atomic proposal to rely on the PCRS for containment appears too premature for acceptance because of the large number of penetrations in the vessel that could cause a serious loss of coolant accident if one of them should fail. For this study, it was assumed that a secondary containment vessel, in accordance with an alternate HTGR design, should be provided.

The SGR plant utilizes equipment much the same type as that demonstrated in the Hallam plant but of larger size. The principal departure from the Hallam design is in the proposed calandria concept for the reactor vessel. The concept appears to bear an element of risk, since a break in one of the 0.025-in. stainless steel tubes could require a prolonged shutdown for repair.

A preliminary analysis of plant availability requirements indicates that a high degree of reliability is required of all reactor equipment, particularly for the HTGR, SGR, and HWR-U concepts, which require more frequent refueling shutdowns. The analysis indicates that each of these plants is presently marginal in meeting a 0.90 plant availability factor, assuming no unscheduled down time for reactor equipment maintenance.

The state of development of the systems and components required for the operation of the reactors evaluated in this study is far from equal.

The light-water-cooled reactors have the most extensive operating experience to call on. Heavy-water reactors have the advantage of NPD, CVTR, PRTR, and HWCTR experience, with the commissioning of the 200-Mw(e) CANDU reactor anticipated soon. The SGR has, as a base, both the SRE [5 Mw(e)] and HNPF [75 Mw(e)] in operation and additional background in sodium technology from Fermi, EBR-1, and EBR-2. Until operations at the Peach Bottom, Dragon, EGCR, and Oldbury stations are under way, the technology of the HTGR must be considered the least firmly established.

References

1. L. G. Bell and W. Evans, Heat Treatment and Mechanical Properties of Zirconium-2.5 wt % Niobium Alloy, Canadian Report CRGM-1035, AECL-1395, Chalk River, Ontario, November 1961.

2. S. N. Tower, 1000 Mwe Supercritical Pressure Nuclear Reactor Plant Study; Selection of Preferred Reactor Concept, USAEC Report WCAP-2042, Westinghouse Electric Corp., July 1962.

3. L. J. Chockie et al., In-Reactor Creep Measurements on Zircaloy-2, Symposium on Radiation Effects on Metals and Neutron Dosimetry, Report ASTM-STP-341, ASTM, 1963.

4. N. E. Hinkle, Effect of Neutron Bombardment on Stress-Rupture Properties of Some Structural Alloys, Symposium on Radiation Effects on Metals and Neutron Dosimetry, Report ASTM-STP-341, ASTM, 1963.

5. V. F. Estcort, Plant Management and Other Factors Affecting Maintenance Costs in Steam Generating Stations, Report ASME-55-A-87, ASME, August 1955.

6. CAPITAL COSTS

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The capital cost estimates derived in this study are based on an evaluation of information submitted by the sponsors for the reference plant designs.[1] Since the amount of detail on design requirements and costs varied widely from one concept to another, it was deemed necessary to normalize certain design requirements to place the concepts on a more equivalent basis for comparison. Features of the plants that were amenable to normalization to a reference standard design, such as the PWR, were discussed in Chapter 5. Methods used in arriving at relative costs of the unique features of the plants were made as consistent as possible to identify the principle cost differences among the concepts. Since the cost normalization methods used are sensitive to the accuracy of the reference design costs, the values reported should be regarded as relative rather than absolute investment costs.

The major portion of the cost normalization was performed by Sargent & Lundy. The procedure used in arriving at the adjusted capital costs consisted of the following steps:*

1. The costs reported by the sponsors were identified in as much detail as possible and tabulated in accordance with the AEC classification of accounts outlined in TID-7025 (Vol. I).
2. From the engineering studies described in Chapter 5, design parameters were revised as necessary to make concepts more comparable with regard to plant performance, equipment requirements, and safety considerations.
3. The costs of systems and components common to more than one plant were normalized to a reference cost base.
4. The costs of systems and components unique to each concept were checked against applicable cost experience and modified when apparent discrepancies were found. (References to the previously published articles by Sargent & Lundy that were used in performing independent cost estimates are listed at the end of this chapter.)

After review of the information reported, the following systems or components were selected for detailed examinations because of their major cost and sensitivity to the particular requirements of a concept:

1. Buildings and containment,
2. Reactor vessel and primary system,
3. Moderator system (HWR),
4. Intermediate heat transfer system (SGR),
5. Feedwater purification system,
6. Feedwater pumps and heaters,
7. Steam, condensate, and feedwater piping,

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*In these discussions, "reported costs" means the cost data supplied by the sponsor and "adjusted costs" means the revised estimates derived from this study.

8. Turbine-generator and circulating-water system,
9. Electrical auxiliaries.

Other systems were either not reported in sufficient detail for estimating purposes or their reported costs were not of significant magnitude relative to the foregoing items to justify detailed cost estimates.

The percentages used in determining indirect costs were reduced from those that would be obtained by using methods outlined in the AEC Cost Handbook, TID-7025 (Vol. I). The indirect cost determinations were based on assumptions made in the ground rules for this study; namely, that the systems of a particular plant had been completely developed and that several plants of that type would be built.

Pertinent modifications and cost estimating procedures used in arriving at the adjusted capital costs of the plant are discussed in Appendix B. The results of the capital cost study are discussed below.

Capital Cost Comparisons

The capital costs derived from this study are compared with the sponsors reported costs in Table 6.1. A summary breakdown of these costs according to the AEC classification of accounts is given in Tables 6.2 and 6.3. Modifications to reference design information for each concept that account for the major portion of the differences between reported and adjusted costs are outlined below.

Table 6.1. Capital Cost Summary

Concept	Direct Costs (millions of dollars)		Total Investment (millions of dollars)	
	Reported	Adjusted	Reported	Adjusted
PWR	104.7	94.3	165.1	133.0
SSCR	81.6	88.1	117.0	124.5
HWR-U	100.7	88.1	151.1	124.7
HWR-Th	108.1	95.7	162.1	135.4
HTGR	81.4	83.4	123.2	117.9
SGR	93.9	93.2	142.9	131.6

PWR

The majority of the direct costs were not changed. The reduction noted in Table 6.1 is accounted for principally in a reduction in the size and weight of the reactor vessel, a change in the primary coolant pump from a canned-motor type to a controlled-leakage type, and application of a multiplier of 0.78 to the turbine price to reflect current markets.

Table 6.2. Summary of Reported Costs

All costs in millions of dollars

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Account						
21 – Structures	12.157	10.674				10.096
22 – Reactor equipment	54.941	37.568				55.692
23 – Turbines	33.246	29.398				24.597
24 – Electrical equipment	3.787	3.130				2.700
25 – Miscellaneous	0.609	0.822				0.857
Total direct cost	104.740	81.592	100.700[a]	108.100[a]	81.379[a]	93.942
Indirect cost						
General and administrative	6.494	5.060				5.198
Miscellaneous construction	1.112	1.190				0.991
Architect-engineer fees	12.358	11.680				14.819
Nuclear-engineer fees	4.500					
Startup	0.600	0.500				0.500
Land	0.360	0.360				0.360
Interest	21.906	10.741				15.542
Contingency	13.017	6.023				11.581
Total indirect cost	60.347	35.454	50.400[a]	54.000[a]	41.861[a]	48.991
Total plant cost	165.087	117.046	151.100	162.100	123.240	142.933
Plant fluids cost		(b)	27.000	33.000	(b)	0.560

[a]Reported costs not in accordance with AEC classification of accounts.

[b]Not included.

Table 6.3. Summary of Adjusted Costs

All costs in millions of dollars

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Account						
21 – Structures	12.209	9.494	10.867	11.346	8.683	9.438
22 – Reactor equipment	50.548	47.116	45.376	52.516	47.479	57.592
23 – Turbines	26.710	26.710	27.044	27.044	22.758	22.322
24 – Electrical equipment	3.986	3.990	4.004	4.004	3.717	3.029
25 – Miscellaneous	0.800	0.800	0.800	0.800	0.800	0.800
Total direct cost	94.253	88.110	88.091	95.710	83.437	93.181
Indirect cost						
General and administrative	5.655	5.287	5.285	5.742	5.006	5.591
Miscellaneous construction	0.990	0.934	0.934	1.015	0.884	0.988
Architect-engineer fees	5.045	4.717	4.716	5.123	4.466	4.988
Nuclear-engineer fees	2.119	1.981	1.981	2.152	1.876	2.095
Startup	0.727	0.770	0.920	0.962	0.707	0.746
Land	0.360	0.360	0.360	0.360	0.360	0.360
Contingency	10.916	10.216	10.229	11.106	9.674	10.795
Interest	12.968	12.137	12.152	13.194	11.492	12.824
Total indirect cost	38.789	36.402	36.577	39.654	34.465	38.387
Total plant cost	133.042	124.512	124.668	135.364	117.902	131.568
Plant fluids cost		6.076	27.000	33.000	0.045	0.560

SSCR

The primary pumps, piping, and heat exchanger prices were adjusted to the same performance and cost base as the PWR, and the turbine estimate for the PWR was used. The lower cost of the SSCR structures compared with the PWR lies in the smaller containment building. Estimated costs for control and instrumentation were increased, since it was not apparent that enough in-core instrumentation had been included in the reported cases.

HWR

The estimated cost of the steam generators was increased to ensure uniformity with the unit cost of the PWR steam generators. Significant reductions were made in the estimate for containment, and the turbine-generator costs were reduced to agree with those of the PWR.

HTGR

The cost of the HTGR circulating-water system was adjusted to correspond with costs for condensers at conditions that were assumed for the other studies. An additional reduction was the result of a new estimate of the turbine-generator cost. Incremental cost adjustments were made for the addition of a secondary containment structure and for internals of the prestressed concrete reactor vessel that resulted in a net increase in HTGR costs, as shown.

SGR

The reported direct costs of the steam and feedwater piping were increased, and the turbine generator costs were decreased to correspond to those of the HTGR.

Cost Uncertainties

Throughout this report, mention has been made of the costs either unreported or not reported in accordance with the AEC classification of accounts for the various reactor concepts. These missing data have, in turn, led to large areas of cost uncertainty in that cost estimates have been prepared for these items, as necessary, in order to permit complete reporting of normalized costs for each concept. In many cases technical information or experience or both upon which to base a cost estimate was almost lacking, while in other instances the intent of the manufacturer's designs was not completely described. In all these cases, assumptions were made for purposes of establishing a cost estimate, and these assumptions and costs are clearly subject to question.

Those areas that constitute the greatest economic uncertainty center on the reactor and primary loop equipment. Other accounts, such as structures and the turbine-generator plant equipment, were adjusted to a normalized base, after which it was found that the adjusted costs did not differ significantly from those reported by the manufacturers. Thus, it is judged that the turbine plant and building estimates, being fairly well understood, are far less subject to error than are the reactor plant equipment estimates.

The specific areas of greatest uncertainty are listed below:

1. The unit cost of the PWR pressure vessel was derived from the costs of a larger, thicker walled vessel originally estimated by Combustion Engineering Company. It is likely that the adjusted unit cost of the PWR vessel is high, since the smaller vessel would not be as difficult to fabricate and install as the larger vessel and therefore its unit cost should be appreciably less.

2. Costs of controlled-leakage pumps were obtained by a differential adjustment of the costs of the canned-motor pumps originally proposed for the PWR. The uncertainty associated with this adjustment would be reflected in the water-cooled plant costs.

3. In normalizing system costs to the PWR, several of the SSCR reported costs were adjusted to higher values. There was not sufficient detail available to judge whether the much lower values reported for the SSCR on heat exchanger costs and control system requirements could be ascribed to the extent indicated to special features of the concept.

4. The reduction in volume of containment structures for the HWR plants was predicated on the design being based on vapor-holding requirements rather than being controlled by equipment layout requirements. This assumption appears to be reasonable but must be checked in detail before the costs assigned can be considered firm.

5. The largest single uncertainty in the HTGR costs rests in the estimates for the prestressed-concrete reactor structure, since no construction experience for a vessel with these requirements is available in the United States.

6. Less experience is available to check the costs of the HTGR helium circulators as compared with experience that can be referred to when considering the coolant circulator costs for the other concepts. Since the HTGR helium circulators require additional development, the manufacturing cost of these units must be considered uncertain at this time.

7. Increases assigned to the steam and condensate piping costs for the SGR could not be checked in detail. Detailed checks may indicate a smaller increase in cost than assigned.

8. Other uncertainties in the SGR adjusted estimate as compared with the HTGR adjusted estimate were:

a. The reactor building requirements and associated costs were much higher than those of the HTGR. It was not possible to normalize these costs as was done for the water-cooled plants because detailed knowledge of layout requirements is lacking.

b. The unit costs of the graphite moderator and reflector for the SGR appear to be higher than those for the HTGR. Since specification requirements may account for this difference, no adjustment was made in the costs.

9. It should be noted that a new cost estimate was received from Atomics International on June 16, 1964 in which the direct costs of the SGR plant were reduced to $\$87.6 \times 10^6$ from the originally reported figure of $\$93.9 \times 10^6$. This information was not received early enough to incorporate in this study. However, the major portion of the cost reduction was based on decreases of approximately 10% for many of the equipment items. AI indicates that the reference SGR report used in this study assumed a first plant and that the design improvements and manufacturing know-how could effect cost reductions of this magnitude for the second plant. It is difficult to appraise the relative effect this assumption may make if applied to the other plants. In the ground rules it was assumed that no credit would be allowed for possible discounts other than those allowed for the turbine-generator units. If these reductions were applied to the SGR alone, the net effect would be to reduce the adjusted direct cost of the SGR plant by approximately $\$5 \times 10^6$ from that given in the comparison.

Indirect Costs

Variations in the methods of computing indirect costs had a major effect on the sponsors estimates of total investment, as seen from an examination of Table 6.1. The largest elements in the indirect costs are engineering fees, allowances for contingencies, and interest during construction. These items account for $23,337,000 of the $48,000,000 difference in the reported investment required for the PWR and the SSCR. In the former case, the procedure specified in the AEC cost evaluation handbook was followed; in the second, a different procedure was followed.

The construction schedules for the two concepts, upon which the interest during construction is based, were also markedly different. This difference, coupled with differences in other costs preceding it, resulted in an estimate of interest during construction for the SSCR of some $11,000,000 less than that for the PWR.

It is recognized that such marked differences are unrealistic and that some of the results obtained by rigorous use of the AEC procedures are likewise unrealistic. In order to achieve a more uniform and more realistic estimate of indirect costs, the following percentages were used (in each case, the percentages were applied to the subtotal of all items preceding the respective item):

General and administrative	6% of direct cost
Miscellaneous construction	1%
Architect-engineer fees	5%
Nuclear-engineer fees	2%
Startup	35% of first year's nonfuel operating and maintenance cost
Contingency	10%
Interest during construction	10.8% (based on 48-month construction schedule in all cases)

Summary

The procedure used in arriving at the total plant investments involved normalizations to reference cost bases in order to identify essential differences in costs of the concepts compared. Therefore, although the costs reported are considered to have significance relative to each other, they should not be regarded as absolute estimates of the total capital costs.

As shown in Table 6.1, the total construction costs derived from this study range from $\$118 \times 10^6$ for the HTGR to $\$135 \times 10^6$ for the HWR-Th. Other total costs (in millions of dollars) are 125 for both the SSCR and the HWR-U, 132 for the SGR, and 133 for the PWR. The magnitudes of these total construction costs depend, of course, on assumptions made on indirect cost allocations.

In previous discussions, many of the uncertainties involved in arriving at appropriate unit cost numbers to use in estimating some of the major system and component costs were noted. In addition, values were assumed for many items that were not defined well enough in the information available to permit independent estimates to be made. Figure 6.1 indicates the ranges of direct construction that are believed appropriate to consider when comparing the capital costs of the several reactor plants. The checkered band indicates a range ascribed to the degree of conservatism used in estimating some of the major items; the crosshatched band indicates a range of error that might be expected from assuming values that could not be checked. For the latter range, 20% of the cost of the undefined system was assumed in arriving at the upper cost.

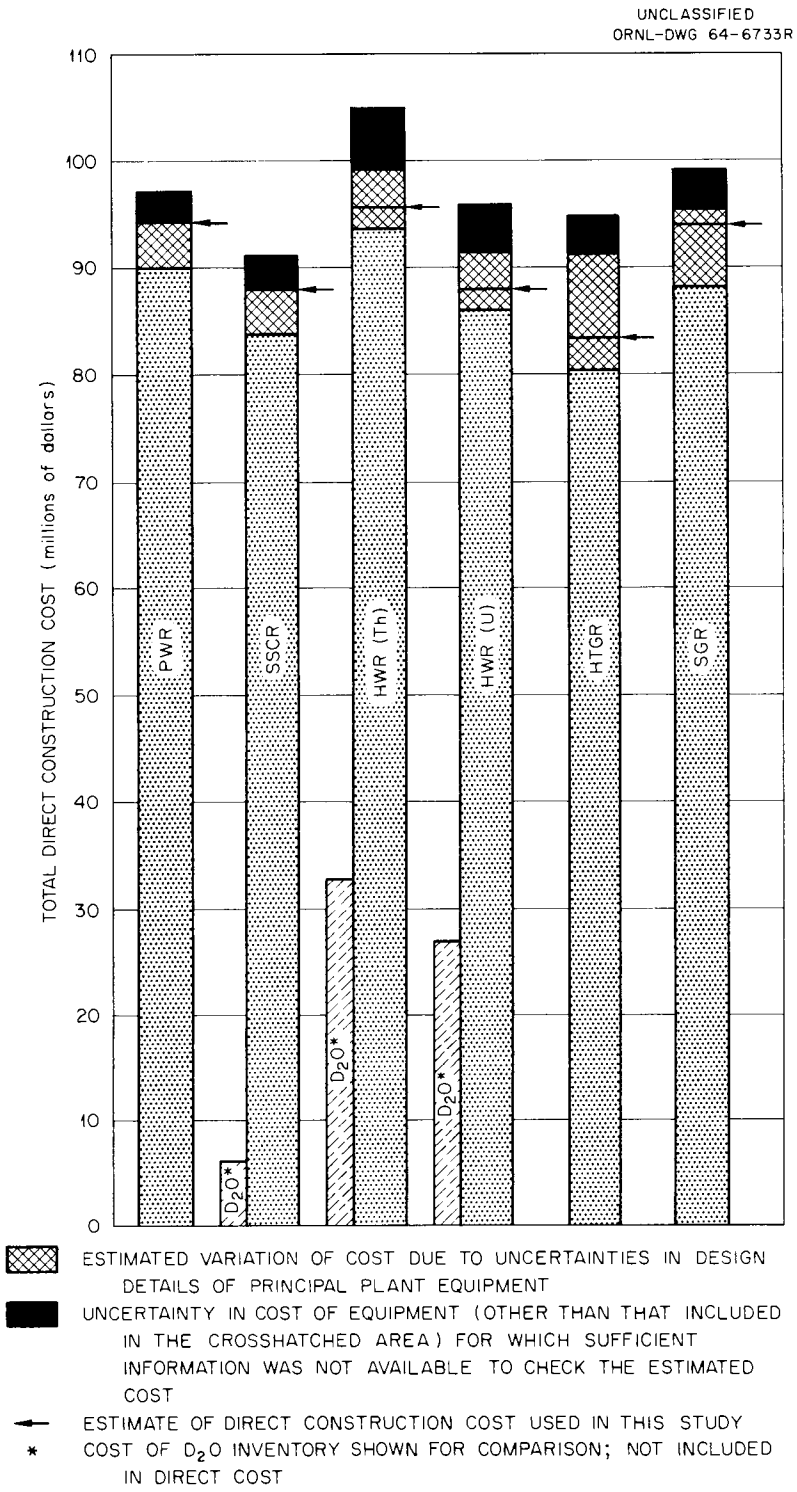

Fig. 6.1. Total Direct Construction Costs Derived from this Study.

Among the water-cooled plants, it may be noted from Fig. 6.1 that the SSCR direct costs are lowest on adjusted totals; they were conservatively estimated in the normalization procedure. The difference of approximately $\$6 \times 10^6$ between the SSCR and PWR can be accounted for by the reduced equipment and containment requirements associated with the smaller reactor vessel of the SSCR.

The HWR-Th plant direct cost is only $\$2 \times 10^6$ higher than the PWR costs, but uncertainties in some of the HWR-Th reactor plant equipment costs and layout requirements could make this difference appreciably more. The HWR-Th direct cost is approximately $\$8 \times 10^6$ higher than the HWR-U cost because of the refueling machine and the larger reactor vessel.

The normalization link between the HTGR and SGR plants and the other concepts is not as secure, although every attempt was made to hold discrepancies in relative costs to a minimum. The supercritical steam system costs of the HTGR and SGR were compared directly with each other and compared with the saturated steam systems. Building and equipment requirements were compared to determine whether differences in costs are consistent with differences in the reference designs.

The HTGR direct cost appears to be some $\$10 \times 10^6$ less than that of the SGR. This is accounted for in the reference designs by features of the "all-inside" HTGR primary system and the fact that the amount of equipment required for the SGR plant is greater due to the need for an intermediate coolant system. The true difference in costs depends heavily on successful demonstration of the HTGR prestressed-concrete reactor vessel with its attendant costs and the development of equipment for the "all-inside" design. If, for some reason, the features of this concept could not be acceptably proven within the allotted time schedule and a more conventional design were used, the costs of the HTGR plant could be appreciably higher.

References

1. See references listed in Chapter 3.

2. Sargent & Lundy, Engineers, Capital Cost Evaluation 1000 MWe Molten Salt Converter Reactor Power Plants, USAEC Report SL-1954, June 29, 1962.

3. Brookhaven National Laboratory, A Review of the Westinghouse 1000 MWe Closed-Cycle Water Reactor Study, May 29, 1963.

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5. Sargent & Lundy, Engineers, 1000 MWe Aqueous Homogeneous Thorium Breeder Reactor Plant Capital Cost Evaluation, USAEC Report SL-1875, June 30, 1961.

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7. Savannah River Laboratory, Heavy Water Moderated Power Reactors Cooled by Liquid D_2O or H_2O, USAEC Report DPW-64-135, March 19, 1964.

8. J. W. Wade, A Computer Program for Economic Studies of Heavy Water Power Reactors, USAEC Report DP-707, Savannah River Laboratory, May 1962.

9. Letter and enclosures from Sidney Siegel, Atomics International to J. A. Lane, Oak Ridge National Laboratory, dated June 16, 1964.

7. OPERATING COSTS

Annual operating and maintenance costs were evaluated for this study by Sargent & Lundy, Engineers. Estimates by the reporting organizations were generally based on a staff of about 70 people, whose salaries and wages amounted to approximately $550,000 per year, including 20% fringe benefits, as shown in Table 7.1. Because of the wide variation in utility manpower policies and the uncertainty involved in determining manpower requirements for sizes and types of plants that have yet to be built and operated, it does not appear to be reasonable to attempt to find differences in manpower requirements for the plants that are reviewed in this study. An annual payroll cost of $554,000 was therefore assumed in all cases.

The cost of maintenance supplies would be expected to vary rather widely with the type of plant, but the amount that could be expected to be spent on items such as cover gas and coolant makeup, special chemical and mechanical treatment expendables, etc. can only be approximately predicted, even with considerable experience in operating a particular plant.

It is to be noted that in the report on the PWR, the basis of the estimate of the annual cost of consumable items was a survey of 29 operating conventional plants, which indicated an annual cost of $1,000,000 for equipment repair and replacement. A portion of this cost would cover control rod replacements. Other consumable materials cost about $100,000 per year. The report also included $71,500 for outside contract services, as required by TID-7025. Thus the total annual cost for consumable materials and services was $1,167,000.

The estimate of annual operating and maintenance costs for the PWR and SGR were realistically prepared, and it appears that the costs are acceptable as a guide to the costs of other systems, subject to distinctions that arise from basic differences in plant designs. The method of arriving at the adjusted costs is summarized in the following.

Total Payroll

PWR payroll figures, including 20% fringe benefits, were applied to all concepts to reflect uniform wage scales, recognizing that approximately 70 people would be required for the operating, maintenance, and supervisory staff.

Insurance

Insurance was based on $60,000,000 of third-party liability insurance at $260,000 per year and federal indemnity at $30 per thermal megawatt per year up to a total coverage of $500,000,000.

Table 7.1. Operating Costs

Reactor Concept	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
As reported						
Number of employees	67	Not reported	Based on TID-7025	Based on TID-7025	Based on TID-7025,	70
Payroll, direct, $/year	461,700	Not reported			extrapolated to	468,838
Fringe benefits	20%	Not reported			1000 Mw	20%
Total payroll, $/year	554,000	Not reported				562,638
Private insurance, $/year	356,600	Not reported				
Federal insurance, $/year			2,100,000	2,100,000		
Repair and maintenance materials, $/year	1,095,500	Not reported				864,400
Repair contract services, $/year	71,500	Not reported				
Makeup fluids, $/year			540,000	660,000	Not reported	109,300
TOTAL COST, $/year	2,077,600	Not reported	2,640,000	2,760,000	1,460,000	1,427,000
As adjusted						
Total payroll, $/year	554,000	554,000	554,000	554,000	554,000	554,000
Private insurance, $/year	260,000	260,000	260,000	260,000	260,000	260,000
Federal insurance, $/year	96,600	97,900	108,000	108,000	68,100	70,500
Repair and maintenance materials, $/year	1,095,500	1,095,500	1,095,500	1,095,500	1,065,000	1,065,000
Contract services, $/year	71,500	71,500	71,500	71,500	71,500	71,500
Fluid makeup,[a] $/year					2,400	109,300
TOTAL, $/year	2,077,600	2,078,900	2,089,000	2,089,000	2,020,000	2,130,300

[a] D_2O makeup charges included under annual charges on D_2O in Chapter 13.

Repair and Maintenance Materials

The repair and maintenance materials cost figure was based on $1,000,000 per year for consumable items and repairs for all plants, plus nominal amounts for supplies peculiar to each concept. All plants have basically the same annual needs of miscellaneous supplies, amounting to approximately $65,000.

In addition, the SGR was charged for makeup sodium, helium, and nitrogen at $109,300 per year, as detailed in the SGR report. The water cooled reactors were charged an additional $30,500 for ion-exchange resin and its shipment. All concepts using D_2O were charged for makeup at 2% of the plant inventory per year, which is included in the cost tabulated in Chapter 13. The HTGR was charged 5% of its helium inventory of 9400 lb for annual makeup. All concepts were charged $71,500 per year for contract services in accordance with the PWR report.

To obtain a true picture of absolute costs, it would be necessary to check against more completely developed designs because of the wide variations possible in several items, notably manpower and repair and maintenance materials. The latter are particularly sensitive to the plant design and complexity. A survey of annual costs of several large conventional plants, all two-unit coal- and gas-fired stations of 500- to 650-Mw capacity, indicates a range of employees from 76 to 159 and annual operating and maintenance costs (other than fuel) varying from $913,000 to $2,350,000, with four of five exceeding $1,900,000 per year. For the plants in this study, it would be reasonable to expect costs of $2,000,000 to $3,000,000 per year, exclusive of fluid makeup.

8. FUEL FABRICATION COSTS*

A. L. Lotts

Fuel fabrication costs have an important bearing on the cost of the fuel cycle for any reactor. Therefore efforts were made to carry out reasonably thorough studies of the cost of fuel fabrication for the various reactors and of the effect of several important variables on such costs.

The fuel element designs furnished by reactor sponsors or manufacturers and described in Chapter 3 of this report were used. All fuel fabrication plant capital and operating costs were included, except the cost of development of the plant site beyond the building structure. Amortization rates of 15, 22, and 30% were applied. It was assumed that fabrication would be done in single-purpose plants producing fuel elements for only one type of reactor; however, some consideration was given to the economics of fabrication in dual-purpose plants. Uncertainties are present in these fuel element fabrication cost comparisons because of the various stages of technological development, lack of fabricating experience at the production rates postulated, and lack of details on some fuel element designs.

The types of fuel fabrication studied are summarized in Table 8.1. For each type, three different amortization rates and two modes of fabrication were considered. In general, costs for ten plant capacities were calculated for each case. The data derived in this study are intended for use at relatively large plant capacities; that is, at production rates required to furnish fuel elements for the power requirements cited. The reference plants are highly automated and accordingly yield a low ratio of operating to capital charges. As plant size is decreased, however, the advantages of automation become less. Finally, at low capacities a point is reached where the product is actually penalized by automation. In the case of such low production rates, the use of more operating personnel and fewer automatic features is advantageous. The exact location of the production rate at which this "law of diminishing returns from automation" applies is obscure. Most probably, the point is really a broad band of production rates. The important consideration in interpreting the data in this chapter is that costs at low capacities are extrapolations of data at higher production rates. Therefore, the costs at low capacities (<1000 kg for all cases except HTGR, <100 kg for HTGR) are probably somewhat high.

Fueling several types of reactors at various stages in their fuel cycle requires the evaluation of fabrication costs obtained under a number of different operating conditions. For example, the SSCR fuel cycle begins with U^{235} and thorium, but the reactor is later fueled with U^{233} and thorium. Metallurgical and ceramic processing of U^{235} with virgin thorium must be done in a closed environment to control alpha activity;

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*This work was done with the advice and assistance of D. A. Douglas, Jr., R. L. Pilloton, and R. B. Pratt.

Table 8.1. Types of Fuel Element Fabrication Studied

Reactor Type	Type of Fuel Element	Fuel	Loading Method
PWR	Rod bundle	UO_2	Vibratory compaction
PWR	Rod bundle	UO_2	Pellets
SSCR	Rod bundle	UO_2-ThO_2	Vibratory compaction
HWR-U	Nested	UO_2	Vibratory compaction
HWR-Th	Nested	UO_2-ThO_2	Vibratory compaction
HTGR	Graphite-virgin BeO	UC_2-ThC_2	Coated particles
HTGR	Graphite-recycle BeO	UC_2-ThC_2	Coated particles
HTGR	Graphite-virgin BeO	UO_2-ThO_2	Coated particles
HTGR	Graphite-recycle BeO	UO_2-ThO_2	Coated particles
SGR	Rod bundle, vented	UC	Slugs
SGR	Rod bundle, nonvented	UC	Slugs

however, with such fuel there is no problem with beta or gamma activity. In the processing of U^{233}-ThO_2, on the other hand, provisions must be made for control of beta and gamma radiation from products of the U^{232} decay chain. Accordingly the three types of plants considered in the study were (1) a direct contact plant, (2) a hooded plant with provision for control of small amounts of alpha activity, and (3) a remote plant capable of handling all recycle fuels.

General Procedure for Estimating

For estimating purposes, costs were broken down into four categories: operating costs, capital costs, fuel element hardware costs, and fuel preparation costs. Each of these was treated separately in the calculations. To enable these estimates to be made, several pieces of basic data were evolved for use in the calculations. These included basic flowsheets, equipment lists, equipment costs at various capacities, and operating data. Costs of hardware, such as end caps, springs, retainers, and general fuel element hardware, were also obtained. In addition, tubing costs were obtained from manufacturers and plotted as a function of wall thickness, diameter, and scale of the production requirement. Basic curves representing the fuel preparation costs as a function of plant capacity for the various types of fuels were prepared.

Other curves were prepared from this basic information to indicate the relation between the following variables and the plant capacity:

Man-hours per 24-hr operating day
Uninstalled cost of various pieces of equipment
Fuel preparation cost
Tubing cost

End cap cost
Spring and retainer cost
Fuel element hardware cost

Equations were written for these various curves and for the corrections to be applied to them in several cases. For example, in the case of the tubing cost, corrections were made for wall thickness and for diameter of the tubing. These equations were then used in computer programs, which were written so that upon identification of the fuel element and upon its description the fuel fabrication costs of the production plant could be computed as a function of capacity. All costs were stated as dollars per kilogram of heavy metal and plant capacities were given in kilograms of heavy metal per operating day. The number of operating days per year was taken to be 260, and the plants were operated on a 24-hr shift.

Engineering Basis

These cost estimates are based on a number of other estimates, both published and unpublished,[1,2] that have been made at the Oak Ridge National Laboratory. Almost all the basic information used in the computer programs was based on the information which had been previously evolved in the prior estimates. In general, those estimates had been made by hand calculation for fuel elements such as the nested tubular fuel element and the rod-bundle type of element (as for the SSCR, PWR, etc.).

Fabrication Flowsheets. The fabrication flowsheets are given in Appendix C. Five flowsheets suffice to explain the processes used for fabrication of the various fuel elements in this study. These are flowsheets for rod bundles with vibratorily compacted fuel rods loaded with pellets, nested tubular fuel elements with vibratorily compacted fuel, rod bundles containing cast UC slugs, and graphite elements containing coated fuel particles.

All these flowsheets are based on what is thought to be conventional practice or what is predicted as the most probable fabrication scheme to be employed in the fabrication of fuel elements now being developed. They are thought to be somewhat conservative in that steps such as autoclaving are included in the flowsheets. It can be argued that, in the future, autoclaving of Zircaloy-clad fuel elements can be eliminated.

Fabrication Plants. The capital costs of fabrication plants were difficult to estimate because insufficient time and funds were available for preparation of actual conceptual designs for the various plants. Thus, the cost of each plant was obtained by multiplying the cost of uninstalled equipment by a factor of 3.45. This relation between total plant cost and uninstalled equipment cost was derived from a study of fuel fabrication costs for large reactors for nuclear desalination purposes,[1] an ORNL study of rod fabrication processes and equipment for the ORNL Kilorod Facility,[3] and data on costs of equipment and plant for the ORNL Thorium-Uranium Fuel Cycle Development Facility (TUFCDF).[4]

In order to obtain the cost of hooded and remote fabrication plants, the costs of the contact plants in this study were multiplied by factors of 1.1 and 2.0, respectively. Thus, total plant costs were obtained by

multiplying the costs of uninstalled plant equipment by the following factors:

Type of Plant	Factor
Contact	3.45
Hooded	3.79
Remote	6.90

The validity of this approach has not been verified over the entire range of capacities and plant types. However, for the purposes of this evaluation it is believed to be adequate for a relative comparison of fabrication costs.

Operating Costs for Fabrication. Operating costs were calculated in a manner similar to that used for obtaining the capital costs. The sources of data for these costs were the same reports and unpublished data. As was the case with the capital costs, data and information which are being derived for TUFCDF were used to determine the manpower to apply. The results of an ORNL Kilorod Facility time and motion study were also used as a basis for estimates of manpower requirements.[5]

For the purpose of the study, the man-year cost was determined to be $28,500. To account for the services to the plants, 10% of the man-year cost was applied for miscellaneous supplies and utilities.

Basic man-hour curves were generated for contact fabrication, and factors were determined that could be used to account for the additional operating labor and overhead expense incurred in the more complicated operating conditions or for operations under more severe circumstances. The problem was therefore similar to that for the calculation of the capital cost. The operating cost factors developed were the following:

Type of Plant	Factor
Contact	1.0
Hooded	1.05
Remote	1.3

Fuel Element Hardware Costs. The fuel element hardware costs were subdivided into several categories. Zircaloy tubing costs were obtained from tubing manufacturers for orders of various size and projected for the technology represented by the period of time being considered in this study. In addition, the tubing manufacturers supplied information concerning the effect of wall thickness and diameter on the cost of the tubing. This information, coupled with information contained in the Guide to Nuclear Power Cost Evaluation,[6] was used to prepare equations for tubing cost in terms of feet of tubing required per day, diameter of tubing, and wall thickness of tubing.

Similar equations or curves were generated for the cost of end caps. To obtain the slopes of the curves, estimates were made of the cost of a particular end cap at a number of different capacities from both small-scale to large-scale production. It was then assumed that machined end

caps required by other fuel elements would scale in a similar way; that is, with the same slope. Therefore, the procedure used in calculating any end cap cost was to determine the cost of the particular end cap at a given capacity. The result was then used in the equation to determine the cost of the end caps at other capacities. The costs of springs, retainers, and other miscellaneous hardware were determined in a manner similar to that used for the end caps. The cost of hardware for the complete fuel element, that is, the rod cage, end fittings, spacers, etc., was determined in a manner similar to that used for the end caps.

Fuel Preparation Costs. The fuel preparation costs were derived from ref. 2. Three different types of fuel were used by the various reactor systems: (1) low enrichment UO_2 was used for the PWR, the HWR, and the SGR; (2) low UO_2, high ThO_2 content was used in the calculation of the fuel preparation cost of the SSCR, the HWR-Th, and the thorium particles for the HTGR; (3) highly enriched UO_2 was used for the uranium particles for the HTGR.

Computer Program for Estimating Fabrication Costs

Because of the large number of variables for which fabrication costs were requested, three computer programs were written to perform the calculations. In these programs, the fuel element is described in terms of the outside diameter of the tubing, the fuel diameter, the length of the fuel rods, the length of the fuel element, the type of fuel, the type of cladding, etc. These data are used as input data for the computer program. In addition, the amortization rates to be applied, the type of fabrication plant to be used, the number of operating days for the plant, and other factors are selected and used as input data. The computer programs then select the proper flowsheet for application to the particular fuel element and compute the various costs; that is, operating, capital, hardware, and fuel preparation costs for that particular case. The results are tabulated as dollars per kilogram vs plant capacity and mills per kilowatt hour vs plant capacity.

Fuel Fabrication Costs

The fuel fabrication costs for the various reactor systems are given in the discussion which follows. For convenience, in the body of this report the fuel costs are presented in the form of curves. Detailed information concerning the data in the curves may be found in ref. 7.

Single-Purpose Fabrication Plants

Fuel fabrication costs were calculated in detail for the cases where only one type of fuel element was to be fabricated in the particular plant. For example, in the first case only the PWR fuel element was to be fabricated. A comparison of nonrecycle and recycle fuel fabrication costs in plants amortized at 22% and producing fuel at the same capacities for various reactor types is given in Tables 8.2 and 8.3. Fuel fabrication costs

Table 8.2. Fabrication Costs for Nonrecycle Fuel in Single-Purpose Plants

Amortization rate: 22%

Reactor	Type of Fuel Element	Fabrication Cost ($/kg of heavy metal) Based on Heavy Metal Throughput[a]					Type of Plant
		500[a]	1000[a]	3000[a]	5000[a]	10,000[a]	
PWR	Vibratorily compacted	114	71	50	45		Contact
PWR	Pellets	127	85	60	53		Contact
SSCR		106	65	45	40		Hooded
HWR-U			47	27.5	23	18	Contact
HWR-Th			52	28	23.5	18.5	Hooded
HTGR	250-μ particles, virgin BeO	162	145				Hooded
HTGR	250-μ particles, recycle BeO	96	82				Hooded
HTGR	1000-μ particles, virgin BeO	164	148				Hooded
HTGR	1000-μ particles, recycle BeO	100	84				Hooded
SGR	Zr clad, vented	100	64	35	28		Contact
SGR	Stainless steel clad, nonvented	96	58	30	23.5		Contact

[a]Heavy metal throughput in kilograms per day.

Table 8.3. Fabrication Costs for Recycle Fuel in Single-Purpose Plants

Amortization rate: 22%

Reactor	Type of Fuel Element	Fabrication Cost ($/kg of heavy metal) Based on Heavy Metal Throughput[a]					Type of Plant
		500[a]	1000[a]	3000[a]	5000[a]	10,000[a]	
PWR	Vibratorily compacted	143	92	68	59		Remote
PWR	Pellets	163	108	77	68		Remote
SSCR		132	80	57	50		Remote
HWR-U			54	32	26	20.5	Remote
HWR-Th			58	32.5	26.5	21	Remote
HTGR	250-μ particles, virgin BeO	180	160				Remote
HTGR	250-μ particles, recycle BeO	115	100				Remote
HTGR	1000-μ particles, virgin BeO	183	165				Remote
HTGR	1000-μ particles, recycle BeO	120	102				Remote
SGR	Zr clad, vented	131	82	44	36		Remote
SGR	Stainless steel clad, nonvented	127	76	39	31.5		Remote

[a]Heavy metal throughput in kilograms per day.

for nonrecycle and recycle fuel in plants amortized at 22% and supplying fuel elements to various reactor types, each type supplying 15,000 Mw(e) of power, are given in Tables 8.4 and 8.5 for the reference exposures.

PWR Fuel Element. Two methods of fuel fabrication were considered for the PWR: (1) vibratory compaction and (2) pelletizing. In all cases, both contact and remote fabrication modes were considered.

The fuel element consists of a closely packed bundle of rods. A few of these rods are slightly smaller in diameter than the others, and, accordingly, an appropriate factor was applied for tubing procurement to account for this difference in geometry. In calculating the fuel costs, the two different diameters were normalized to one value. The outside diameter of the larger rods was 0.378 in. and the smaller rods, 0.320 in.

Table 8.4. Fabrication Costs for Nonrecycle Fuel in Single-Purpose Plants Supporting a 15,000-Mw(e) Power Industry

Amortization rate: 22%

Reactor	Type of Fuel Element	Burnup (Mwd/kg of U)	Plant Capacity (kg of metal per day)	Fuel Cost ($/kg of heavy metal)
PWR	Vibratorily compacted	25	2170	55
PWR	Pellets	25	2170	66
SSCR		30	1800	52
HWR-U		15.4	4080	24.5
HWR-Th		22	2860	29.5
HTGR	250-μ particles, virgin BeO	51.4	716	152
HTGR	250-μ particles, recycle BeO	51.4	716	88
HTGR	1000-μ particles, virgin BeO	51.4	716	155
HTGR	1000-μ particles, recycle BeO	51.4	716	90
SGR	Zr clad, vented	35.0	1100	60
SGR	Stainless steel clad, non-vented	35.0	1100	54

Table 8.5. Fabrication Costs for Recycle Fuel in Single-Purpose Plants Supporting a 15,000-Mw(e) Power Industry

Amortization rate: 22%

Reactor	Type of Fuel Element	Burnup (Mwd/kg of U)	Plant Capacity (kg of metal per day)	Fuel Cost ($/kg of heavy metal)
PWR	Vibratorily compacted	25	2170	70
PWR	Pellets	25	2170	84
SSCR		30	1800	65
HWR-U		15.4	4080	27.5
HWR-Th		22	2860	33
HTGR	250-μ particles, virgin BeO	51.4	716	170
HTGR	250-μ particles, recycle BeO	51.4	716	106
HTGR	1000-μ particles, virgin BeO	51.4	716	176
HTGR	1000-μ particles, recycle BeO	51.4	716	110
SGR	Zr clad, vented	35	1100	78
SGR	Stainless steel clad, non-vented	35	1100	71

In the normalization, the average pellet diameter was calculated to be 0.322 in. The average outside diameter of the fuel rod was 0.369 in. These and other data concerning the fuel element were used in the computer code.

Results of the calculations are given in Figs. 8.1 through 8.4. It may be observed in these figures that the fabrication of PWR fuel by vibratory compaction could result in savings of $7 to $10 per kilogram of heavy metal in comparison with pelletizing.

SSCR Fuel Element. The SSCR fuel element is a rod-bundle type of element. All fuel rods are of the same diameter, and the element presents no particular problems from the standpoint of the geometry.

The fuel is fabricated by vibratory compaction. (The details of the procedure are given in Appendix C.) It was assumed that the plant would operate in sequence on fuel of particular enrichments, and therefore it would be necessary to clean up the equipment when changing enrichments. The problem was to estimate the amount of time required to do the cleanup and to start fuel of a new enrichment through the plant. A 10% turnaround penalty was assigned to account for the enrichment changes.

Fuel element fabrication costs for the SSCR in hooded and remote plants are presented in Figs. 8.5 and 8.6. The costs are approximately $10/kg lower than the costs of the PWR vibration-compacted fuel elements, in spite of the fact that the SSCR fuel is fabricated in a hooded plant and the PWR fuel is fabricated in a contact plant. There are two reasons for this. First, the PWR was penalized for the two rod geometries and the SSCR for the fact that a number of different enrichments were used, but since these penalties are not altogether offsetting, they account for some of the difference. Second, the SSCR fuel rods are 0.420 in. in diameter, whereas the PWR rods averaged 0.369 in. in diameter, which means that the SSCR fuel elements have a greater amount of fuel and fertile material per unit length of tubing than the PWR elements. Thus in most of the plant, it would be expected that the SSCR costs would be less than the PWR costs by a factor proportional to the ratios of the squares of the diameters of the fuel rods being fabricated.

HWR Fuel Elements. The fuel elements for the HWR are of an entirely different type from those discussed above. All the previous elements are of the rod-bundle type, whereas the elements for the HWR are of a nested tubular type.

The fuel element consists of three annuli spaced by ribs and by end fittings on the fuel annuli. Each fuel annulus consists of two concentric fuel tubes connected by the end cap, which is in the form of a ring. Accordingly, each fuel element requires six different sizes of tubing, all of which are relatively large in diameter.

The vibratory compaction procedure is similar to that for the rod-bundle elements, except for the different geometries. (The fabrication flowsheet is shown in Appendix C.) It was assumed that a vibratorily compacted element would perform adequately, and costs were calculated on this basis. Should higher fuel density be required an incremental cost amounting to possibly 5 to 10% would be incurred for swaging. In the fabrication procedure all the bulk material is crushed, ground, and sized in common equipment. However, because of the differences in geometries, three parallel equipment lines are used for fabrication of the

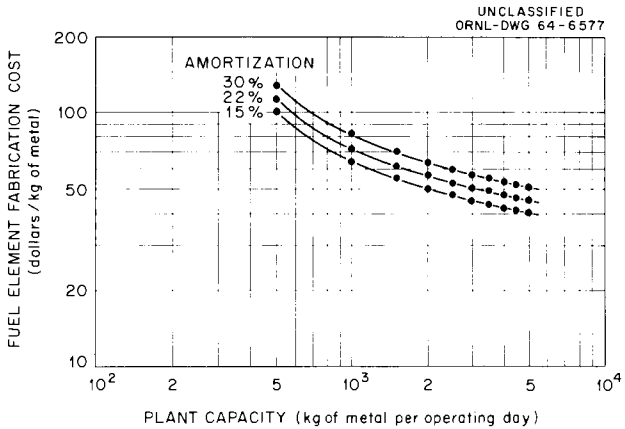

Fig. 8.1. PWR Fuel Element Costs for Fabrication by Vibratory Compaction of the Fuel in a Contact Plant.

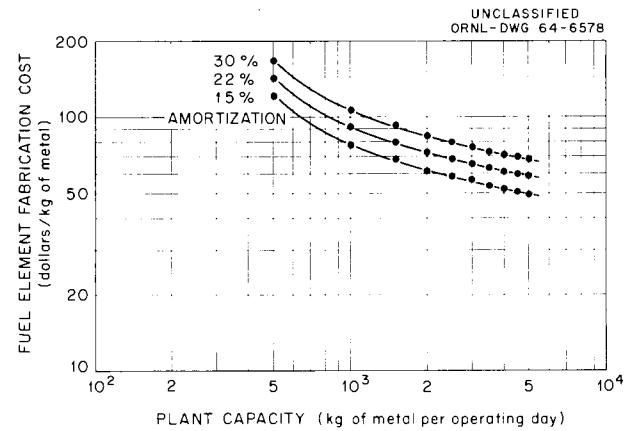

Fig. 8.2. PWR Fuel Element Costs for Fabrication by Vibratory Compaction of the Fuel in a Remote Plant.

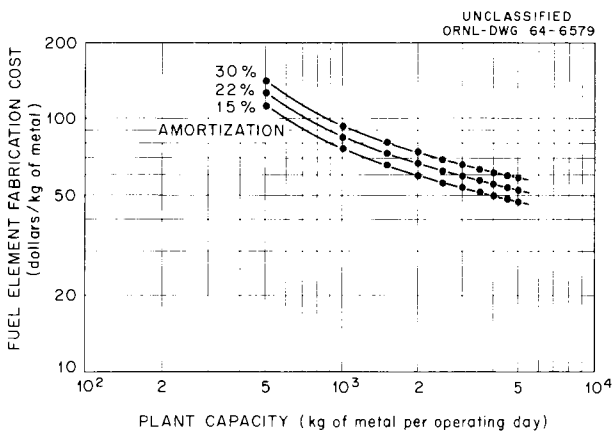

Fig. 8.3. PWR Fuel Element Costs for Fabrication by Pelletizing the Fuel in a Contact Plant.

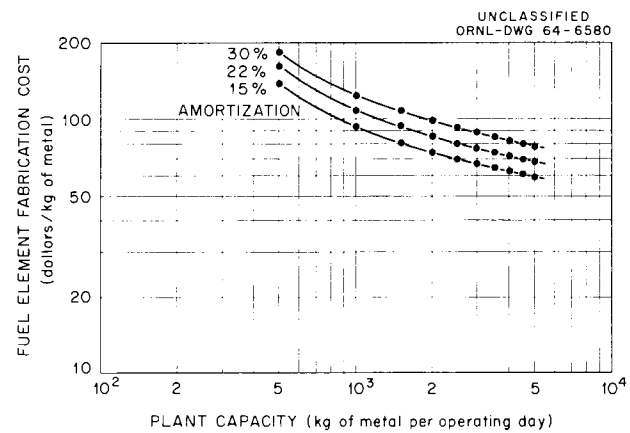

Fig. 8.4. PWR Fuel Element Costs for Fabrication by Pelletizing the Fuel in a Remote Plant.

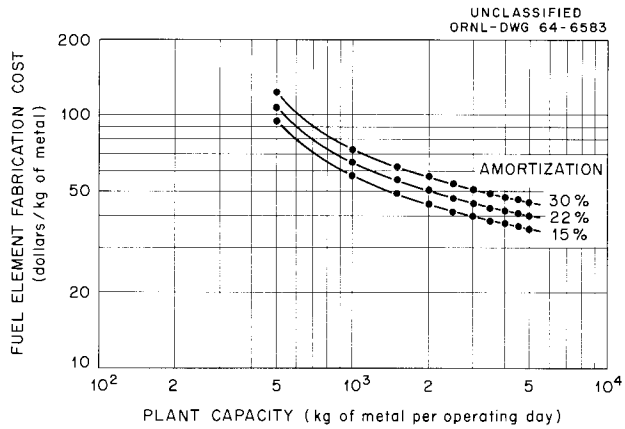

Fig. 8.5. SSCR Fuel Element Costs for Fabrication in a Hooded Plant.

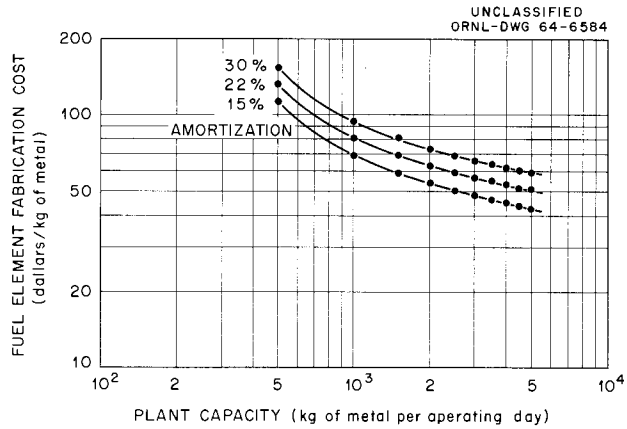

Fig. 8.6. SSCR Fuel Element Costs for Fabrication in a Remote Plant.

three fuel tubes. The final assembly of these fuel tubes is carried out in one production line.

The cost of processing the bulk material for the tubular fuel elements was considered to be equal to that for the rod bundles. The various costs involved in fabricating one fuel tube were calculated, and these costs were multiplied by 3 to obtain the cost of a three-annuli nested tubular element.

Fuel element fabrication costs for the UO_2-fueled HWR reactor are given in Figs. 8.7 and 8.8, and fuel fabrication costs for the ThO_2-UO_2-fueled HWR are given in Figs. 8.9 and 8.10. In comparing these fuel costs with those for reactors that use the rod-bundle type of element, it is observed that these costs are considerably lower. The principal reason for this is that the nested tubular element has a much greater weight of fuel per unit length of tubing. The cost of large-diameter tubing is quite high, but not sufficiently high to offset the advantage that this fuel element design has in its higher loading per unit length of tubing.

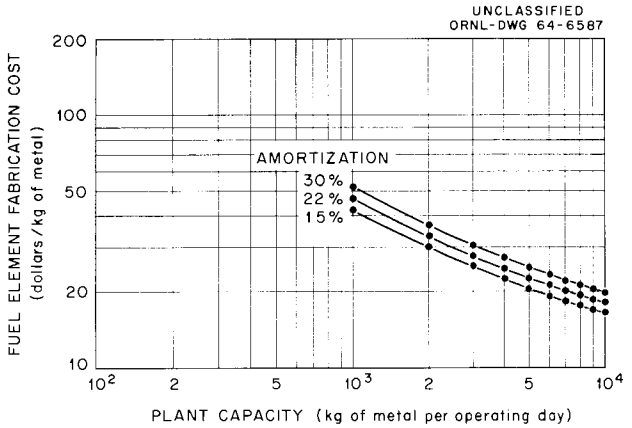

Fig. 8.7. HWR-U Fuel Element Costs for Fabrication in a Contact Plant.

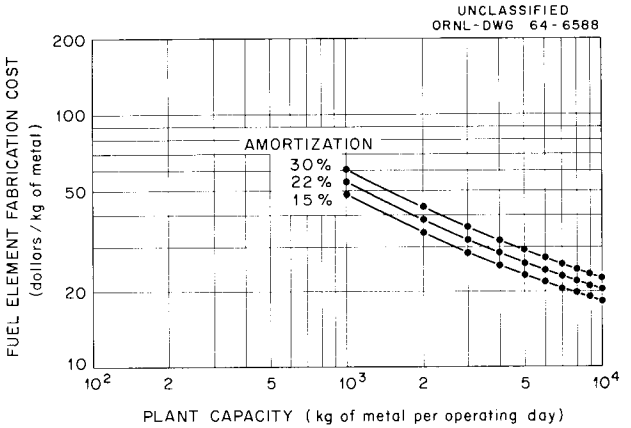

Fig. 8.8. HWR-U Fuel Element Costs for Fabrication in a Remote Plant.

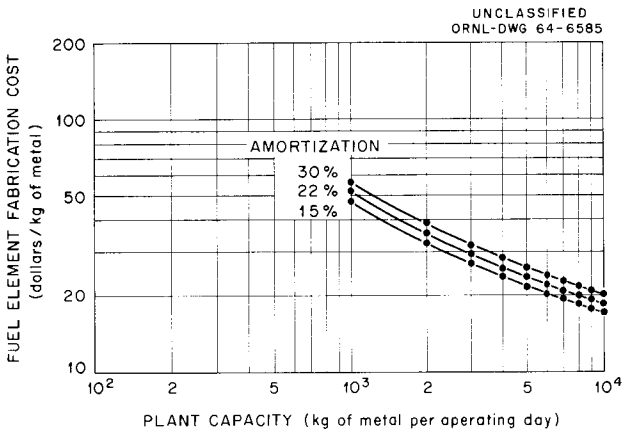

Fig. 8.9. HWR-Th Fuel Element Costs for Fabrication in a Hooded Plant.

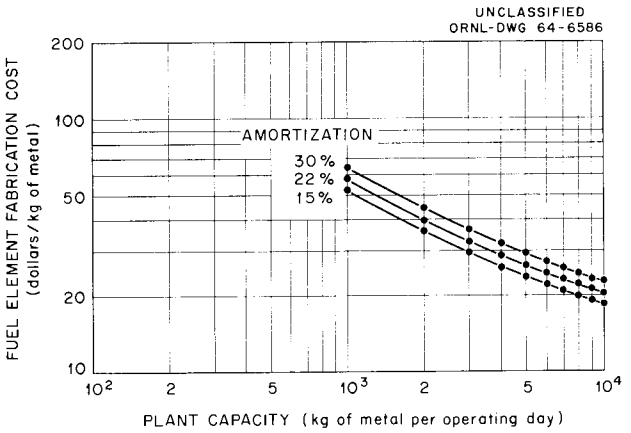

Fig. 8.10. HWR-Th Fuel Element Costs for Fabrication in a Remote Plant.

HTGR Fuel Elements. The estimation of fuel fabrication costs for the HTGR is difficult because of the limited amount of data and experience with this fuel element. In fact, the fuel element is still being developed and has never been fabricated on any scale.

The design of the HTGR fuel element was described in Chapter 3. Since all the data are not given in Appendix C for this element as they are for the other reactors, some of the parameters used in determining the fuel cost are listed below:

Fuel weight per element, kg of U + Th	16.57
BeO weight per element, kg	48.57
Diameter of fuel element, in.	4.7

For the purpose of this cost estimate, it was assumed that spheres of $(Th\text{-}U)C_2$ would be made batch-wise by heating microspheres of thorium-uranium oxide plus carbon in a continuous furnace. It was also assumed that the carbide spheres would be coated with carbon in fluidized-bed coaters that would be larger but basically similar to existing coating furnaces. Improvements in coating technology would probably result in appreciable reductions of the capital charges and the operating expenses.

The method of calculation was essentially the same as for the other elements, but a different computer program was required. The various factors for capital and operating costs were derived and used in the same way.

The hardware costs for the HTGR element account for a substantial part of the total cost. Table 8.6 is a tabulation of HTGR hardware costs that were used at various production rates. The cost of the graphite was derived from various sources. It was based on the assumption that machined AGOT or similar graphite would be acceptable and that tolerances of the order of at least 0.020 in. would be satisfactory for the location of the holes. If the permeability data and the dimensional tolerances initially indicated by the proponent of this reactor could not be relaxed to those assumed for this study, the cost of the graphite part would be at least doubled.

Basic BeO cost data were obtained from commercial sources and organizations using substantial quantities of BeO, and the cost of virgin BeO bodies for the fuel element reflect these data. For the recycle BeO cases,

Table 8.6. HTGR Basic Hardware Costs

Plant production rate, kg of heavy metal per operating day	10	100	500	1000
Graphite cost, $/fuel element	300	290	270	230
Virgin BeO, $/kg of BeO	48	38	32	30
Recycle BeO, $/kg of BeO	14	11.10	9.50	8.80

it was assumed that the irradiated BeO could be ground to a powder, degassed by high-temperature treatment, and repressed and sintered to form a body suitable for use in the recycle fuel elements.

The cost of fabrication of the HTGR fuel elements is given in Figs. 8.11 through 8.18. Figures 8.11 and 8.12 give the fabrication cost if the average particle size is 250 μ and virgin BeO is used. Figures 8.13 and 8.14 are for 250-μ particles and recycled BeO. Figures 8.15 and 8.16 give the fuel cost if the average particle size is 1000 μ and virgin BeO is used. Figures 8.17 and 8.18 are for 1000-μ particles and recycled BeO. As can be seen from these curves, the use of 1000-μ particles will give a greater fuel cost than the use of 250-μ particles. Also, it is indicated that the use of recycled BeO would markedly reduce the fuel cost.

The effect of the larger particles on coating efficiency is not accurately known. Problems could be encountered in the decreased surface area per kilogram of fuel, the much increased gas velocity required to fluidize the particles, the poor gas-solid contact and particle motion characteristics, and the increased maintenance due to sooting. The coefficient used in the computer program to adjust operating and capital costs for various particle sizes was calculated from the following relationship:

$$\text{CK (or AUT)} = \left(\frac{D_0}{250}\right)^{0.1} ,$$

where

CK = capital coefficient,
AUT = operating coefficient, and
D_0 = particle diameter in microns.

This study has also indicated that a saving of 7% of the total fuel fabrication cost could be realized by using coated oxide particles instead of carbide particles.

SGR Fuel Elements. Two types of fuel elements, a vented Zircaloy rod bundle and a nonvented stainless steel rod bundle, were considered for SGR. Both elements would be fueled with hyperstoichiometric UC. The procedure for fabrication is given in Appendix C.

The method of calculation was quite similar to that for the other rod-bundle elements, except that different equations were used to reflect the cost of preparation of UC from UO_2 and for the cost of making the slugs and loading them and sodium into the fuel rods. It was assumed that technological growth would make the preparation of UC and UC slugs an automatic operation. The effect of this assumption was to give a lower cost at the high production rates than if much manual operation were required.

The estimates for the fabrication of the Zircaloy-clad vented fuel elements in a contact plant are given in Fig. 8.19. The contact plant was considered because only uranium is employed in the fabrication of these elements. The cost of fabrication for the same element in a remote plant is given in Fig. 8.20. The cost in contact and remote plants for the nonvented stainless steel-clad element is given in Figs. 8.21 and 8.22. In comparing these figures with those obtained for the PWR, the SSCR, and

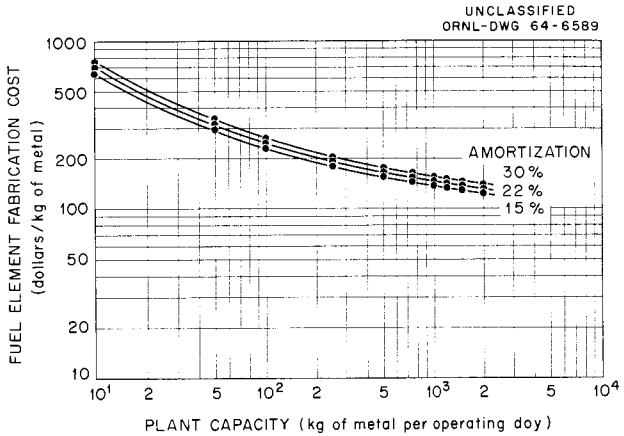

Fig. 8.11. HTGR Fuel Element Costs Based on 250-μ Particles and Virgin BeO Fabricated in a Hooded Plant.

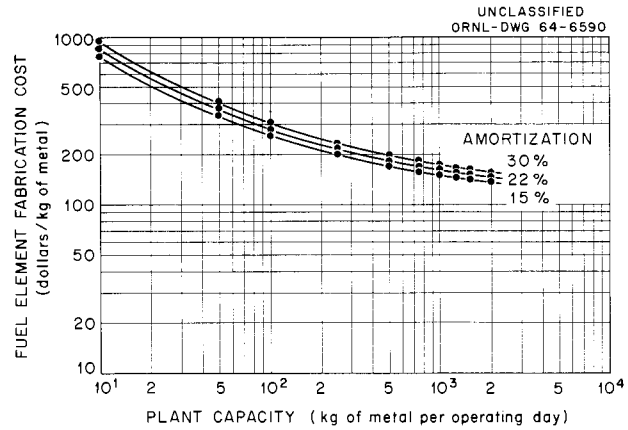

Fig. 8.12. HTGR Fuel Element Costs Based on 250-μ Particles and Virgin BeO Fabricated in a Remote Plant.

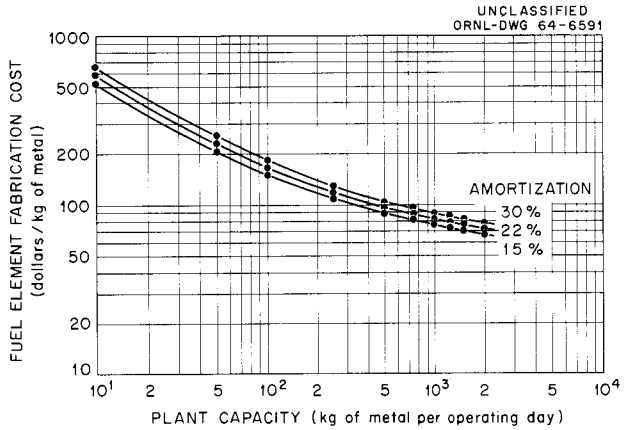

Fig. 8.13. HTGR Fuel Element Costs Based on 250-μ Particles and Recycled BeO Fabricated in a Hooded Plant.

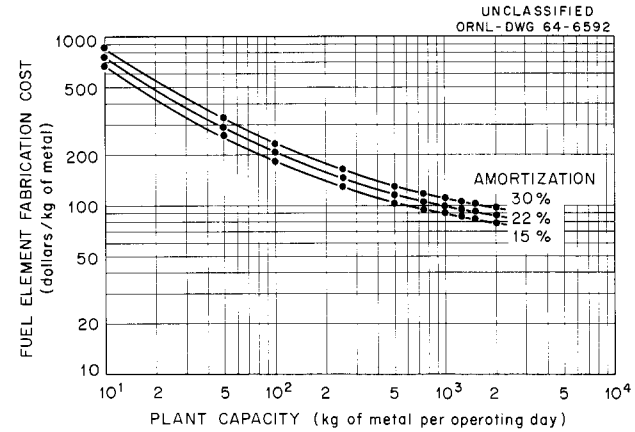

Fig. 8.14. HTGR Fuel Element Costs Based on 250-μ Particles and Recycled BeO Fabricated in a Remote Plant.

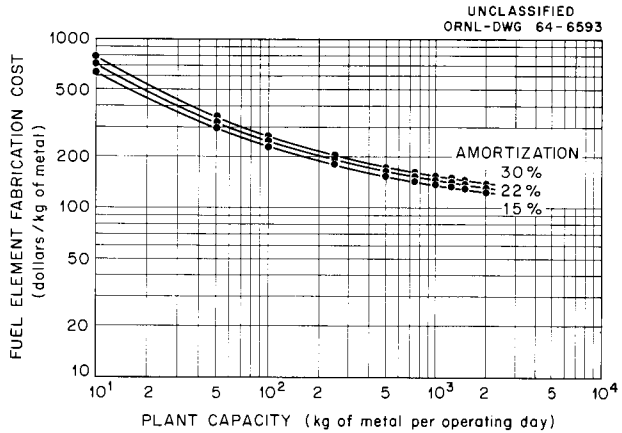

Fig. 8.15. HTGR Fuel Element Costs Based on 1000-μ Particles and Virgin BeO Fabricated in a Hooded Plant.

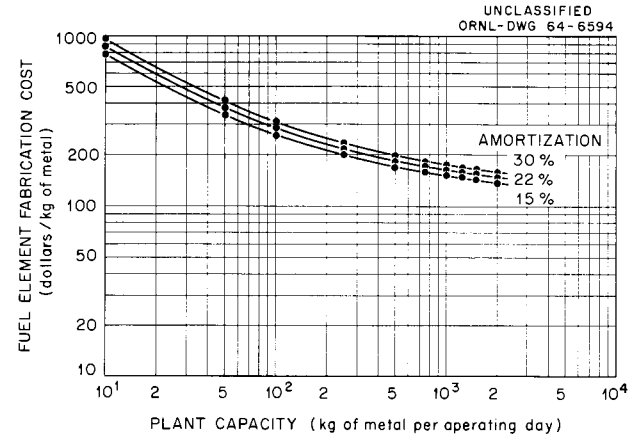

Fig. 8.16. HTGR Fuel Element Costs Based on 1000-μ Particles and Virgin BeO Fabricated in a Remote Plant.

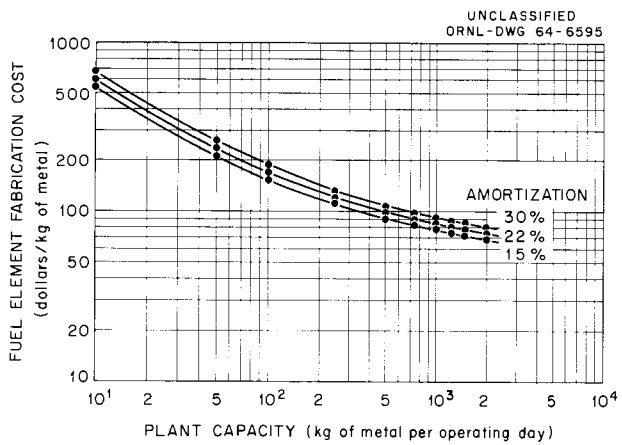

Fig. 8.17. HTGR Fuel Element Costs Based on 1000-μ Particles and Recycled BeO Fabricated in a Hooded Plant.

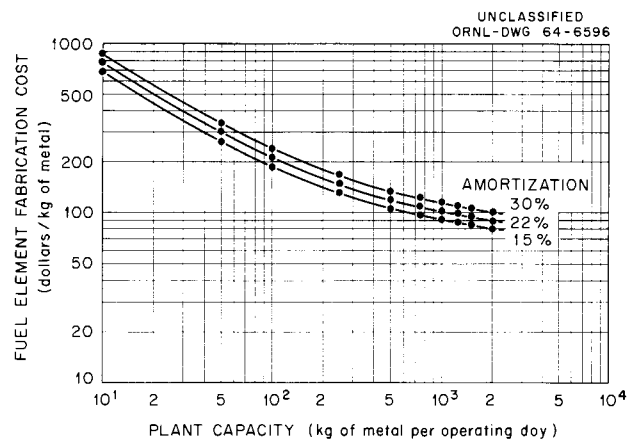

Fig. 8.18. HTGR Fuel Element Costs Based on 1000-μ Particles and Recycled BeO Fabricated in a Remote Plant.

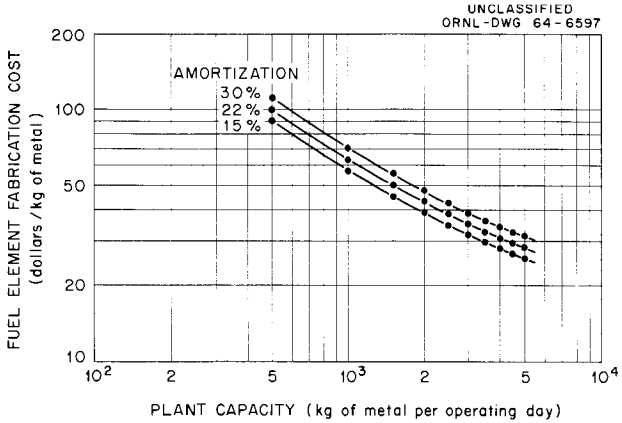

Fig. 8.19. SGR Fuel Element Costs Based on Zircaloy-Clad Vented Rods Fabricated in a Contact Plant.

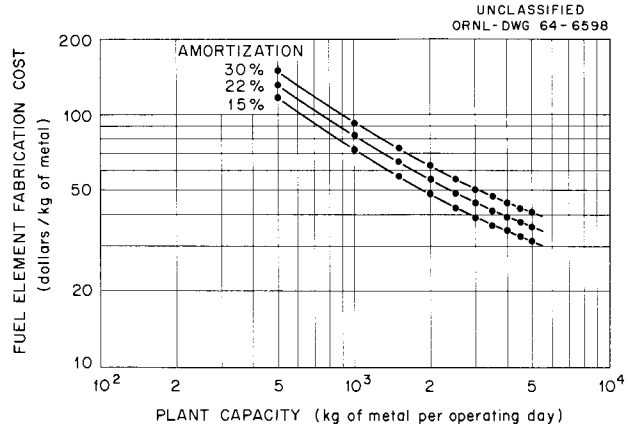

Fig. 8.20. SGR Fuel Element Costs Based on Zircaloy-Clad Vented Rods Fabricated in a Remote Plant.

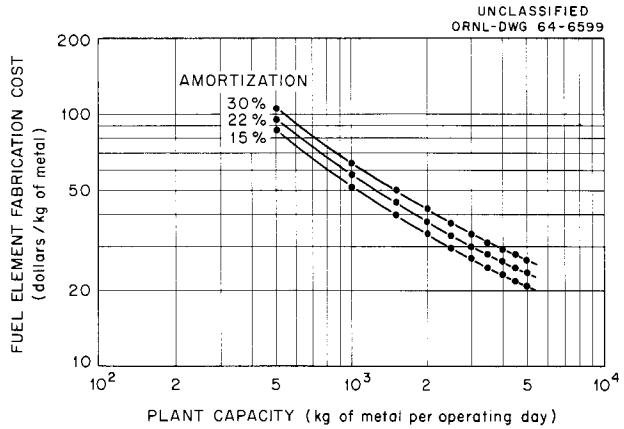

Fig. 8.21. SGR Fuel Element Costs Based on Stainless Steel-Clad Nonvented Rods Fabricated in a Contact Plant.

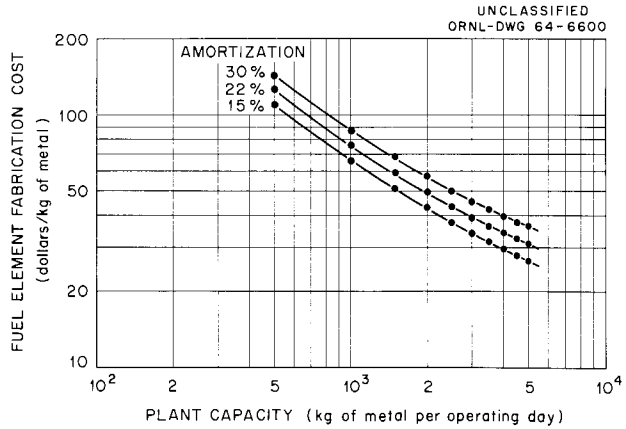

Fig. 8.22. SGR Fuel Element Costs Based on Stainless Steel-Clad Nonvented Rods Fabricated in a Remote Plant.

the SBR, it may be noted that the costs are comparable and sometimes lower for the SGR. The principal reason for this is that the SGR fuel rod is large in diameter. For example, the UC slug diameter for the SGR is 1/2 in. This should be compared with an average pellet diameter for the PWR of 0.322 in. or a fuel diameter of 0.369 in. for the SSCR. In addition, the fuel compound UC is approximately 95% uranium by weight, and the density of UC is greater than that of UO_2. These factors result in an SGR fuel element with a much higher weight of fuel per unit length of tubing. In addition, no penalties are imposed for turnaround time due to different enrichments or different sizes of tubing.

Dual-Purpose Fabrication Plants

Calculations of fuel-fabrication costs in dual-purpose plants were not done in detail, but the problems of operation in dual-purpose plants were given some attention in the form of engineering judgment. In all the plants, some advantage could be gained because of the combination of overhead functions, such as some of the administration. In low-capacity plants this would be a much greater factor than in large-capacity plants. The saving on the administrative charges at low-capacity plants, for example, 500 kg/day, might be of the order of 10%; whereas at large-capacity plants, 5% could probably be saved. These savings on administrative charges would apply to all the cases studied. Additional savings that would result with the combination of two different types of fuel elements or a fuel element of a certain type with a PWR element are considered in the discussion that follows.

SSCR-PWR Fuel Elements. In fabricating SSCR and PWR fuel elements in a dual-purpose plant, it appears that some saving in equipment and operating cost could be realized at low capacities, possibly below 2000 kg/day. If vibratorily compacted PWR and SSCR fuel were being fabricated, all operations would have to be carried out in hoods because the SSCR fuel contains thorium oxide. The cost of the UO_2 PWR fuel would therefore be penalized to some extent in order to accommodate the SSCR. In addition, throughout most of the range of production rates investigated (even below 2000 kg/day) in the rod fabrication phases of the operations more than one equipment unit is required. Therefore, little economy would result because equipment units would not be eliminated. The two fuel elements do, however, require a number of pieces of equipment which are the same, and therefore it appears that the saving in equipment and operating cost within the plant and some advantages in procurement of hardward would result in possibly 5 to 10% cost savings at the low capacity. It is thought that a saving of 10% could be realized at a capacity of 500 kg/day, 5% at 1000 kg/day, and 0% at 2000 kg/day.

HWR-PWR Fuel Elements. There is a radical difference between the design for the HWR fuel element, which consists of nested tubes, and the PWR element, which is a rod bundle. This means that not much of the equipment could be commonly employed. It is visualized that only the powder-conditioning operations could be combined in any sense. A further point is that the powder-conditioning operations account for only approximately 8% of the total fuel fabrication cost for the PWR if vibratory compaction is used; this figure would be somewhat greater for the HWR elements.

Therefore, if the powder conditioning is the only operation which can be combined, no large saving could be expected. At a capacity of 500 kg/day, the saving could be as high as 4%; at 1000 kg/day, 2%; and at 2000 kg/day, 0%.

SGR-PWR Fuel Elements. The only two common attributes in the SGR and PWR fuel elements are the common feed material (UO_2) and the type of cladding used. However, the type of UO_2 would probably be quite different. Also, the fuel rods have different diameters and are loaded differently. Thus, it is questionable that any of the rod fabrication equipment could be employed for both types of fuel elements. Accordingly, it is indicated that the cost saving obtained by combining fabrication of these two elements would be very little. At 500 kg/day, a saving of 2% might be realized; at 1000 kg/day, 1%; and at 2000 kg/day, 0%.

HTGR-PWR Fuel Elements. The fuel elements for the HTGR and for the PWR have only one common characteristic, that is, the feed material, which is UO_2. It is probable, however, that the feed material will be radically different; that is, for the HTGR, the fuel would be spheres of UO_2 and, for the PWR, fragmented shards, if vibratory compaction is used, or a ceramic-grade UO_2, if pelletizing is used. Therefore, it is highly questionable that any advantage could be realized by combining any of the operations for fabrication of these two fuel elements, and therefore no saving could be realized.

Summary

Data are presented in Table 8.7 that illustrate the factors which account for the differences between the fuel costs of the various reactors evaluated. The values listed for the amount of heavy metal per unit length of tubing for the various fuel elements are particularly significant. There is a substantial correlation with the fuel cost. The length of the fuel rods also affects costs in that a number of processing steps, such as end capping and inspection of end-cap welds, are independent of length; the unit cost for end capping a 2-ft fuel rod is approximately the same as for a 10-ft rod, although the rod would have only 1/5 as much fuel. Expressed in dollars per kilogram, the cost of end capping the 2-ft rod would be 5 times the cost of end capping the 10-ft rod.

The fuel element fabrication cost is strongly sensitive to: (1) the diameter of the fuel rods or the weight of heavy metal per unit length of tubing or cladding; (2) the fuel density of the fuel element, that is, the amount of uranium per unit volume of loaded fuel tube; (3) the number of fuel element geometries in a reactor (the greater the number of geometries the greater the cost); (4) the number of enrichments employed in the reactor (the greater the number of enrichments the more frequent is the shutdown of the fabrication plant and, therefore, the higher the cost); and (5) the fuel enrichment itself, which, if it is sufficiently high, will limit the capacity of the equipment.

Table 8.7. Comparison of Fuel Element Parameters and Fuel Element Costs

	PWR		SSCR	HWR-U	HWR-Th	HTGR (1000-μ particles, virgin BeO)	SGR (Zircaloy-clad vented rods)
	Vibratorily Compacted Fuel	Pelletized Fuel					
Fuel type	UO_2	UO_2	UO_2-ThO_2	UO_2	UO_2-ThO_2	UC_2-ThC_2	UC
Fuel dimensions, in.							
Outside diameter	0.326	0.322	0.369				0.500
Length	132.0	132.0	135.0	188.0	188.0		174.0
Fuel density (theoretical), g/cm^3	10.95	10.95	10.04	10.95	10.02		12.97
Fuel density attained, % of theoretical	88	96.5	88	88	88		100
Fuel rods per assembly	234	234	238				18
Fuel rod (or tube) dimensions, in.							
Outside diameter	0.369	0.369	0.420	(a)	(a)		0.590
Wall thickness	0.0215	0.0215	0.0255	0.025	0.025		0.020
Fuel element autoclaved	Yes	Yes	Yes	Yes	Yes	(b)	No
Amount of metal per unit length of tubing, kg/ft	0.158	0.169	0.185	1.48	1.36		0.500
Cost of fabrication of nonrecycled fuel at a rate of 1000 kg/day, $/kg	71	85	65	47	52	148	64

[a] Large diameter elements with three fuel annuli per element.

[b] Not applicable.

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5. J. J. Varagona, A. L. Lotts, and J. E. Van Cleve, Jr., Time Study of Fuel Rod Fabrication in the Kilorod Facility (to be published).

6. Kaiser Engineers, Guide to Nuclear Cost Evaluation, Vol. 4, Fuel Cycle Costs, March 15, 1962.

7. Communication from A. L. Lotts, Oak Ridge National Laboratory, to M. W. Rosenthal, August 15, 1964.

9. SPENT FUEL PROCESSING COSTS

J. T. Roberts*

From 1957 to 1963 the standard basis for evaluating the spent-fuel processing contribution to nuclear power cost was the calculated cost for processing the fuel in question in the "AEC Reference Fuel-Processing Plant,"[1] a hypothetical plant capable of recovering purified uranium and plutonium from irradiated fuel at the rate of 1000 kg of uranium of up to 3% enrichment per day. At higher enrichments the capacity of this conceptual plant decreased, as a result of criticality considerations, to, for example, 930 kg/day at 4% enrichment, 537 kg/day at 10%, and 44 kg/day at 93%. The reference plant could process thorium fuels at the rate of 1000 kg/day if only the enriched uranium were to be recovered or 600 kg/day if thorium also had to be recovered, again subject to criticality limitations on the enriched-uranium processing rate. The AEC announced in the Federal Register of March 12, 1957 that it would provide spent-fuel processing services at calculated charges based on the conceptual plant "on an interim basis ... until the time when processing is available commercially." Initially the standard AEC daily charge was $15,300, both for the calculated number of processing days required for a batch of fuel and for the calculated number of "turnaround" days (the time required between processing batches for shutdown, cleanout, and startup), but there was a provision for escalation that increased this figure to more than $17,000 in 1961 (ref. 2) and to an estimated $19,800 by 1965 (ref. 3). The charges on a per-metric-ton basis were typically 25 to 100% or more higher than on the per-day basis, depending on batch size and daily processing rate.

In 1963 the AEC accepted an offer by Nuclear Fuel Services, Inc., to provide fuel processing services on a commercial basis, beginning in 1965 (ref. 3). Nuclear Fuel Services (NFS) is now building a plant with a nominal capacity of 1000 kg/day for uranium of up to 3% enrichment irradiated to burnups of up to 20,000 Mwd/MT and lower capacities for higher enrichments and burnups. The nominal capacity for thorium fuels is 500 kg/day for recovering only the enriched uranium. An extra charge will be assessed for the disposal of the thorium-bearing waste. The initial base charge of $23,500 per "revenue day" (processing plus turnaround time) is subject to future escalation. Since the minimum turnaround time under the NFS formula is one third the processing time, the minimum per-metric-ton price for processing is $31,300. For 3% enriched fuel irradiated to a burnup of 20,000 Mwd/MT at a thermal efficiency of 31%, this corresponds to a minimum processing cost contribution to nuclear power of about 0.21 mills/kwhr(e), with typical costs higher:

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*This work was done with the advice and assistance of R. Salmon, F. E. Harrington, J. W. Ullmann, D. E. Ferguson, F. L. Culler, J. O. Blomeke, W. G. Stockdale, and others in the ORNL Chemical Technology Division.

	Typical Costs [mills/kwhr(e)]
Southern California Edison	0.31
Indian Point UO_2	0.36
Indian Point ThO_2	0.75

Above 3% enrichment the NFS processing rate penalties are somewhat greater than the AEC conceptual plant penalties; for example, 880 kg/day at 4%, 465 kg/day at 10%, and 40 kg/day at 93%.

Both the AEC conceptual plant and the NFS actual plant were somewhat arbitrarily sized, with neither of them large enough to service a nuclear power economy of the size hypothesized for the advanced-converter evaluation. At its nominal capacity, NFS could service a maximum of less than a 5000-Mw(e) nuclear power industry. Also, the AEC and NFS plants were not designed to process all types of fuel considered in the evaluation; head-end treatments for SGR and HTGR fuels are not included. Therefore it was decided to estimate processing plants that would fit the particular amounts and types of fuels to be used in the hypothetical nuclear economy being evaluated. Under the ground rules adopted, each type of fuel was considered separately for the case of processing in a single-purpose plant designed to service reactors with a capacity of 15,000 Mw(e), each reactor being of 1000-Mw(e) capacity. As an alternative, an estimate was also made of the case of a dual-purpose processing plant designed to service pressurized-water reactors with a capacity of 10,000 Mw(e) plus reactors of the type under consideration, also with a capacity of 10,000 Mw(e). Capital and operating costs were estimated for each hypothetical processing plant.

Cost Bases and Assumptions

The most important assumption for these estimates was that the Du Pont cost study by Farrow[4] should be used as a basis for the detailed cost calculations, as it had been for previous estimates of costs of fuel processing for a large nuclear water-desalination industry.[5-9] The Du Pont study was primarily a detailed design and cost estimate for a radiochemical separations plant for processing daily 10 short tons of aluminum-clad natural uranium irradiated to a burnup of 5000 Mwd/ton. It was assumed that the fuel contained 4 kg of plutonium per ton and that it had cooled 180 days after reactor discharge. The processing plant was considered to have an on-stream capability of 85% (310 days per year), and it was designed for remote maintenance. A secondary estimate was made of the changes which would be made if the scale were reduced to 1 ton per day. Other secondary estimates were made for direct maintenance and for processing Zircaloy- or stainless steel-clad slightly enriched uranium after a chemical-decladding head-end treatment.

For the present study the basic 1- and 10-tons/day estimates[4] were escalated by approximately 12% to reflect 1964–1965 price levels and modified to reflect differences in head-end treatment and fertile, fissile,

and fission-product material throughput rates required for the various advanced-converter fuels. Costs not included in the Du Pont study were added to cover land, startup, and ultimate disposal of radioactive wastes.

The Du Pont cost estimates are thought to be realistic by comparison with experience in large government-operated plants and accurate to within perhaps ±20% under the assumptions made. They are, however, based on a more conservative philosophy than the NFS estimates, as was pointed out by the committee of technical consultants who reviewed the NFS proposal for the AEC.[3] The Du Pont 1-ton/day estimate, as modified for the present evaluation, is equivalent to an increase in NFS capital and operating costs on the order of 50% or more. Most of the difference in operating costs is accounted for by a factor of 2 difference in personnel. A detailed comparison of the difference in capital costs is not yet available. In recognition of the lower NFS predicted costs, the present estimates were based on the detailed Du Pont estimate, but the final results were reduced by 20% to the approximate lower limit of the Du Pont estimate. This amounts to a judgment that the NFS estimate still seems optimistically low but that competitive pressures may produce actual costs lower than the Du Pont estimates.

Capital Investment

The Du Pont capital cost estimates totaled $\$43 \times 10^6$ and $\$60 \times 10^6$ for the basic 1- and 10-tons/day plants, respectively. Escalation to 1964–1965 price levels increased these figures to approximately $\$48 \times 10^6$ and $\$67 \times 10^6$, respectively. The Du Pont estimate was broken down into 24 categories. For the present evaluation these were combined into four categories:

1. Those items assumed to scale with the gross annual fuel throughput,
2. Those items assumed to scale with the gross daily fuel throughput (dependent on annual throughput and also on turnaround time),
3. Those items assumed to scale with the daily throughput of highly enriched uranium or plutonium,
4. Those items assumed to scale with annual fission product throughput.

The items in the first two categories were much larger than those in the last two categories, but the differences between fuels was enough to be of some significance in the cost comparisons in the last two categories. In general the thorium-containing fuels can be processed through dissolution and solvent extraction equipment of a given size only at a rate of 50 to 60% of the uranium processing rate through the same equipment.[1,10] This difference was taken into account in scaling the second category of cost items, but it did not amount to a very severe penalty against thorium fuels because of the low scaling factor. In the dual-purpose plants the processing rates of the two materials were matched as closely as practical, but matching was not possible in all cost categories, and the larger of the two costs was controlling.

A head-end capital cost was added, as appropriate, for the types and amounts of fuel, and some of the dual-purpose plants required two sets of

head-end equipment. For the Zircaloy-clad oxide fuels, a mechanical shear-leach head-end treatment was assumed, and the cost was calculated from a previous estimate that a 2.66-MT/day head end would cost $\$4.1 \times 10^6$ and would scale approximately as the 0.5 power.[11] This was assumed to scale with the fuel throughput in metric tons per day, though obviously the fuel element design would also be important. The SBR seed and blanket rods were assumed to be sheared and leached together, since they would both be present in the same assemblies; no penalty was assessed for the difficult-to-dissolve UO_2–ZrO_2 seed on the assumption that either a satisfactory dissolution process would be developed or a different seed material would be used. For SGR fuel, a shear-pyrohydrolysis-leach head-end treatment was assumed.[12] A rough approximation of the right order of magnitude indicated that this treatment would cost 50% more than shear-leach alone. For HTGR fuel the head-end treatment was assumed to consist of cutting open the element (or shipping tube), pouring out the fuel particles, separating the particles into two size fractions, burning the large particles, dissolving the resulting oxides, and transferring the small particles to ultimate waste disposal containers or, alternatively, recovering the uranium in them separately. The cost of this HTGR head end was assumed to be equal to that of an equivalent size shear-leach head end. This rough approximation is not thought to be in error enough to affect significantly the calculation of total capital investment.

A capital cost for off-gas treatment for iodine and rare-gas removal was added based on $\$1 \times 10^6$ for a 1-ton/day plant and scaling with the 0.6 power of the nominal daily fuel throughput. This is consistent with previous estimates, but it does not reflect a detailed design study.

Up to this point the calculated total capital costs were equivalent to a modification of the Du Pont "total project cost."[4] In order to obtain the total capital investment, however, it was necessary to add land, startup, and working capital costs, which were considered beyond the scope of the Du Pont estimates.

Any land cost estimate must be arbitrary until a particular geographical location is specified; however, the same problem exists in making standard reactor capital investment estimates, and the AEC "Guide to Nuclear Power Cost Evaluation"[2] has provided ground rules for estimating a cost for comparison purposes. The example calculations give a land cost that varies (very roughly) with the 0.6 power of the reactor power. In the processing plant case it was decided to vary the land cost with the 0.6 power of the annual fission-product throughput, assuming the reported $500,000 NFS land cost to be appropriate for a plant processing 225 metric tons per year of fuel irradiated to a burnup of 20,000 Mwd/MT. For the single-purpose plants in the present study the calculated land cost varied only over the range $\$0.7 \times 10^6$ to $\$1.0 \times 10^6$. It is not implied that this is a highly significant difference, only that the possible difference in exclusion area requirements between plants processing different amounts of fission products should be recognized.

The AEC "Guide"[2] specifies capitalizing startup costs equivalent to 35% of one year's operating and maintenance cost. This guide was intended for reactor cost estimations, but it does not seem unreasonable to apply it also to a processing plant. The "Guide" calculations of working capital requirements are somewhat more complicated, but when applied to the

Du Pont estimates they were approximately equivalent to 19% of the annual operating cost. For the present evaluation, startup and working capital were calculated together on the basis of 54% of the annual operating cost (see below). As mentioned earlier, the total was "adjusted" downward by 20%.

Fixed-Charge Rates on Capital Investment

The ground rules for the present evaluation specified a reference fixed-charge rate of 22% for calculating annual capital charges from total capital investment for a privately owned processing plant, with alternative rates of 15 and 30%. These fixed-charge rates are two to four times higher than the 7.7% figure used in earlier estimates for desalination reactor processing plants based on municipal financing. The difference is highly significant in the calculated total cost, since the ratio of capital investment to annual operating cost is in the range of 10 to 12.

The 7.7% rate corresponds roughly to a 30-year plant life financed 100% with 4% interest debt, with no federal income tax, but with some allowance for interim replacements, insurance, and other taxes; it is typical for a municipally financed power plant.[13] It might be appropriate to a large municipal water-desalination industry but not to a privately owned nuclear utility industry unless the government should decide to do "toll reprocessing" at the Hanford or Savannah River processing plants.

The 15% rate is equivalent to the privately owned utility financing rate usually assumed in AEC reactor cost calculations, that is, something like a 30-year life, 50% bond financing at 5%, and 50% stock financing at 8.5%, plus federal corporation income taxes, other taxes, interim replacements, and property and liability insurance.[13] It would normally be considered an unusually favorable fixed-charge rate for a privately owned processing plant, since chemical companies normally depreciate their plants over periods much shorter than 30 years, have a much lower bond-to-stock capital ratio than 50/50, and expect to earn much more on capital stock than 8.5%. Interestingly enough, however, though all the details have not been made known, the NFS price schedule appears to reflect the equivalent of approximately a 15% fixed-charge rate and has been said to cover amortization over a 15-year period.[3] This apparently is made possible by a high debt-to-equity ratio, approximately 75/25, at moderate interest rates.

A 30% fixed-charge rate would be typical for a company financed essentially 100% with common stock equity capital, the depreciation of the plant over 10 years or less, and the expectation of earning 12% or more after taxes. Such a company, however, usually has a ratio of total capital investment to annual operating cost on the order of unity, so a high fixed-charge rate on capital does not unduly burden the selling price of the product. This is certainly not the case for spent fuel reprocessing plants, and it is doubtful that the spent fuel processing industry will grow along this line, assuming that the NFS venture is successful.

The 22% rate adopted as a reference value was a compromise. It was recognized that future processors might not be able to arrange financing

as favorable as that of NFS and that competitive pressures in a growing industry might force depreciation of plant cost in less than 15 years.

Annual Operating Costs

The Du Pont estimates[4] for the basic 1- and 10-short-tons-per-day plants were $\$3.75 \times 10^6$ and $\$6.23 \times 10^6$ per year, respectively, not including the costs of operating off-gas treatment facilities or head-end facilities of the type assumed in the present evaluation. After correcting for these additions and escalating to 1964–1965 price levels, annual operating costs for the advanced-converter processing plants were assumed to scale with the same relationship between operating and plants costs as that indicated for the two Du Pont plants. In other words, the total plant cost (not including land, startup expense, and working capital) was estimated first, and the annual operating cost was estimated by comparison with the Du Pont capital and operating costs. Thus the four cost-category variables mentioned in the discussion of capital costs were allowed to influence the operating cost also, but indirectly rather than by a detailed calculation. Again, the original estimate was adjusted downward by 20%.

Ultimate Waste Disposal Charge

The Du Pont estimate[4] allowed only for what was considered a reasonable amount of waste storage tanks and spares. No provision was made for building new tanks or for eventual replacement of tanks or other ultimate disposition of the fission-product wastes. For the purposes of the present evaluation, it was assumed that the Du Pont costs adequately covered ultimate disposal of low- and intermediate-level wastes but provided only for a minimum of interim storage of high-level liquid wastes. It was decided to add a separate annual charge to provide the equivalent of a sinking fund for the eventual ultimate disposal of the high-level wastes. The charge was based on cost estimates[14] developed over the past few years for an ultimate disposal scheme consisting of

1. Interim liquid storage, preferably in the as-evaporated acidic form rather than neutralized,
2. Calcination to the thermally stable solid form in stainless steel pots that would serve as the ultimate disposal vessels,
3. Interim on-site storage of the pots to permit further reduction in heat generation rate,
4. Shipment of the pots to a salt-mine disposal facility,
5. Ultimate disposal of the pots by burial in a deep salt mine designed to dissipate the decay heat by conduction to the earth.

The disposal cost has not been finally optimized, but for the particular 15,000-Mw(e) nuclear economy studied in some detail it appears that the total cost need not exceed 0.03 mills/kwhr(e). Rather than apply this figure as a flat charge in the present evaluation, it was decided to scale it with the total annual fission-product throughput to the 0.6 power, since for burnups above about 10,000 Mwd/MT for Purex and Acid-Thorex flowsheet acid wastes, the fission products are controlling rather than the inert material. Empirically, this annual charge can be expressed

as $0.643(\text{MTFP/year})^{0.6}$ in units of millions of dollars per year (MTFP = metric tons of fission products). This charge should not be applied blindly to any case, but it is thought to provide a reasonable basis of comparison for the present evaluation. It is conservatively high by comparison with the NFS-New York State infinite tank storage charge,[3] but it represents a more sophisticated ultimate disposal scheme. Further optimization may produce a lower cost estimate than the approximate 0.03 mills/kwhr(e), and for the present study this estimate also was adjusted downward by 20%.

Plant Sizing

The fuel types, compositions, burnup levels, thermal efficiencies, specific powers, and batch sizes were specified by other groups of the evaluation team. In many cases preliminary estimates were used in the absence of final calculations, but the differences are not expected to cause significant errors in the fuel processing cost estimates. Because the various groups were doing their calculations simultaneously, the processing costs reported in this chapter may not always agree exactly with those shown in other chapters.

As described previously in this chapter, the primary cost-determining variables were gross annual fuel throughput rate, gross daily fuel throughput rate, gross daily highly enriched fuel throughput rate, and gross annual fission-product throughput rate. The relationship between annual and daily rates depends on the split of the 310 operating days per year between processing days and turnaround days. The "conventional" turnaround formula was used. It specifies that for a given fuel batch the number of turnaround days is equal to the number of processing days, with a minimum of 2 and a maximum of 8 turnaround days per batch.[1,3] The additional NFS specification that turnaround days may not be less than one-third of the processing days was not applied because it was not thought to be appropriate to a processing plant designed specifically to match a given fuel load of only a few separate batches per year. The turnaround formula used is probably too conservative, since it was not obvious that a processing plant receiving many similar batches per year should spend an equal time on turnaround and processing as the formula requires for 19 to 77 batches per year. The requirement is not oppressive, however, because of the low scaling factors typical of these cost estimates. At a scaling factor of 0.2, typical of this study, the difference between zero turnaround and equal turnaround and processing times is only 15%, and at least some penalty against reactors turning out many small batches would seem to be justified for administrative and other reasons.

Burnup and thermal efficiency did not influence the cost estimates as such but entered them in the form of the four throughput-rate cost categories. Specific power did not influence the estimates, since its effects were assumed to be washed out after 150 or more days cooling before processing. The 150-day cooling time requirement is that applied by NFS.[3] The Du Pont study[4] assumed 180 days but did not provide iodine scrubbing of off-gases. A reduction to 120 days at little or no increase in cost might be possible, but it was not studied. The shipping cost optimization studies (Chapter 10) did not indicate a significant cost advantage in 120-day processing.

No enrichment penalty was applied to any of the fuels except in the case of separately recovering the HTGR highly enriched uranium (which was alternatively assumed to be discarded or dissolved together with the thorium). The average enrichment of the spent fuels was generally less than or only slightly greater than 3% fissile material, at which level it was assumed to be possible to control the criticality hazard by administrative procedures or by poisoning of the process equipment with fixed or solution poisons. In the special case of the separated HTGR uranium, 40 kg/day of fissile isotopes was assumed to be equivalent to a nominal 1 MT/day of low-enrichment uranium. Overall, this amounts to a somewhat more "liberal" criticality philosophy than applied in the AEC[1] and NFS[3] cost formulas and considerably more than the Du Pont[4] basis, but it is thought to be reasonable in a plant designed to operate on repetitive batches of similar fuel. Because of the low scaling factors, a conventional enrichment penalty would have imposed only a modest extra cost burden on a few of the fuel types, hence the assumption did not make as large a difference as it would, for example, under the NFS enrichment penalty formula.

Miscellaneous Costs

By the ground rules, losses in chemical processing were assumed to be 1%. They were not included in the cost totals for processing as normally reported but were added separately in arriving at the fuel-cycle cost.

Likewise, use charges on the processing plant inventory are normally itemized separately from the processing cost as such. For this evaluation the processing holdup time was assumed to be 30 days (for pre- and post-processing holdup) plus the number of processing days per batch. The total time from reactor discharge to completion of processing is given in Chapter 10 by the shipping cost optimization studies.

Conversion and reconversion costs are also normally kept separate from spent fuel processing costs, though in the nuclear fuel economy envisioned in the present evaluation, they might well be done on-site in the processing plant or the adjacent fabrication plant. The processing costs reported here are for products in the form of nitrate solutions. The plutonium is assumed to be sold in this form. The thorium and its associated highly enriched uranium are assumed to be transferred to the fabrication plant in this form to be converted into recycle fuel, except in the special case of the separately recovered HTGR uranium, which is assumed to be sold as the nitrate. For the low-enrichment uranium that must be recycled through gaseous diffusion, a reconversion charge to cover converting the uranyl nitrate to UF_6 must be added separately. The AEC flat charge[2] of $5.60/kg could be applied, but in a large private economy reconversion might be done on-site by or in connection with the processing plant. In this case a sliding (log-log) scale from $5.60 at 1 MT/day to $2.70 at 10 MT/day might be more appropriate.

The costs of conversion of fresh and makeup uranium from UF_6 to oxide or other forms required for fuel fabrication are discussed in Chapter 8. In this evaluation, in general, the spent uranium from the HWR-U reactors is of the same order of enrichment in U^{235} as diffusion plant

tails; hence the expense of reconverting it to UF_6 is not justified. In this case a small penalty probably should be assessed for conversion to oxide and on-site disposal or storage.[5]

In the HWR-U cases, in general, and in some of the PWR cases, depending on the plutonium price assigned, a throw-away fuel cycle should be considered, since the value of the recovered fuel is less than the cost of processing (some similar instances may occur in the reconversion of depleted UNH to UF_6 when the cost of reconversion is greater than the value of the resulting UF_6). This possibility has been neglected in this evaluation by the ground rules, but substitution of some sort of "indefinite storage" charge (in anticipation that the cost of processing may decrease with time) for the shipping and processing charges might be considered. The AEC "Guide"[2] suggests using $5/kg in this case, and some other studies have used numbers as low as $1 to $2/kg.[7,8]

Cost Estimates

The calculated costs are listed in Table 9.1 for single-purpose plants processing fuel from units of a single reactor type producing 15,000 Mw(e). The costs were calculated for the indicated "reference" conditions of burnup, which were believed to represent near-optimum conditions from the point of view of total power cost. Table 9.2 presents similar cost calculations for "off-reference" conditions to establish the variation of processing costs with burnup or throughput rate in the assumed constant total production nuclear economy. The "total capital investment" was multiplied by the appropriate fractional fixed-charge rate and added to the "annual operating cost" and "annual ultimate disposal charge" to give the total annual and unit costs listed at the bottom of the table. It can be seen that doubling the fixed-charge rate from 15 to 30% increases the annual and unit costs about 50%.

The unit costs in dollars per metric ton processed vary widely over the range of burnups and throughputs covered by this study, but the total annual costs and the unit costs in mills per electrical kilowatt-hour vary over a relatively narrow range because of the low cost-scaling factor typical of these fuel processing plants and because the economic optimization of the systems has tended to "push" the reference reactors to burnup values that correspond to almost the same processing costs. Figure 9.1 shows unit costs per ton of fuel processed as a function of annual fuel throughput. It emphasizes how similar all the fuels are if compared at the same processing rate. Figure 9.2 shows unit costs per kilowatt-hour as a function of burnup. It tends to emphasize the differences between reactor types more than the tables or Fig. 9.1, but it points out how little reduction in processing cost results from large changes in burnup under the condition of constant total power. Figure 9.3 shows the capital investments, annual operating costs, and ultimate waste disposal charges from which the unit costs were developed.

The HTGR cases in Tables 9.1 and 9.2 are for the same fuel handled in two different ways. In Table 9.1 the thorium and uranium particles are assumed to be separated and recovered separately so that the uranium can be resold. In Table 9.2 it is also assumed that the particles are

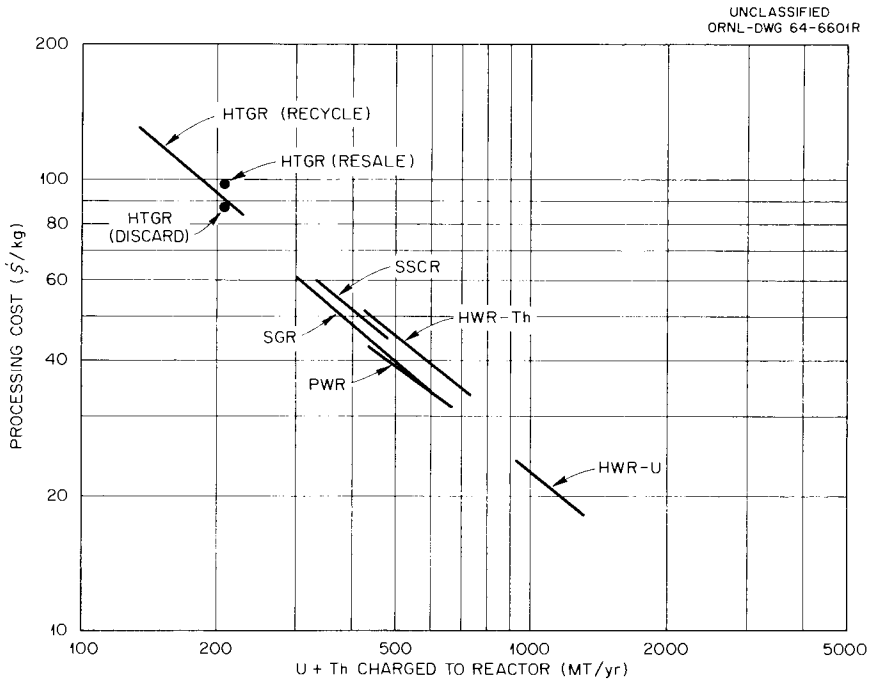

Fig. 9.1. Processing Cost as a Function of Throughput in Single-Purpose Plants. Based on a 22% fixed-charge rate.

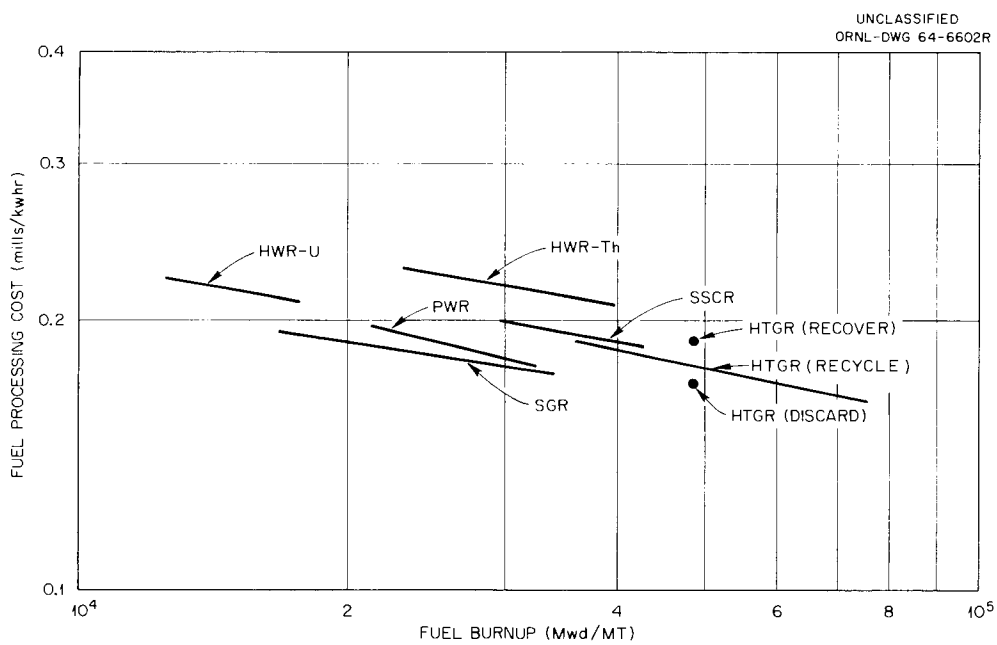

Fig. 9.2. Fuel Processing Contribution to Power Cost as a Function of Burnup for 15,000-Mw(e) Single-Purpose Plants with a 22% Fixed-Charge Rate.

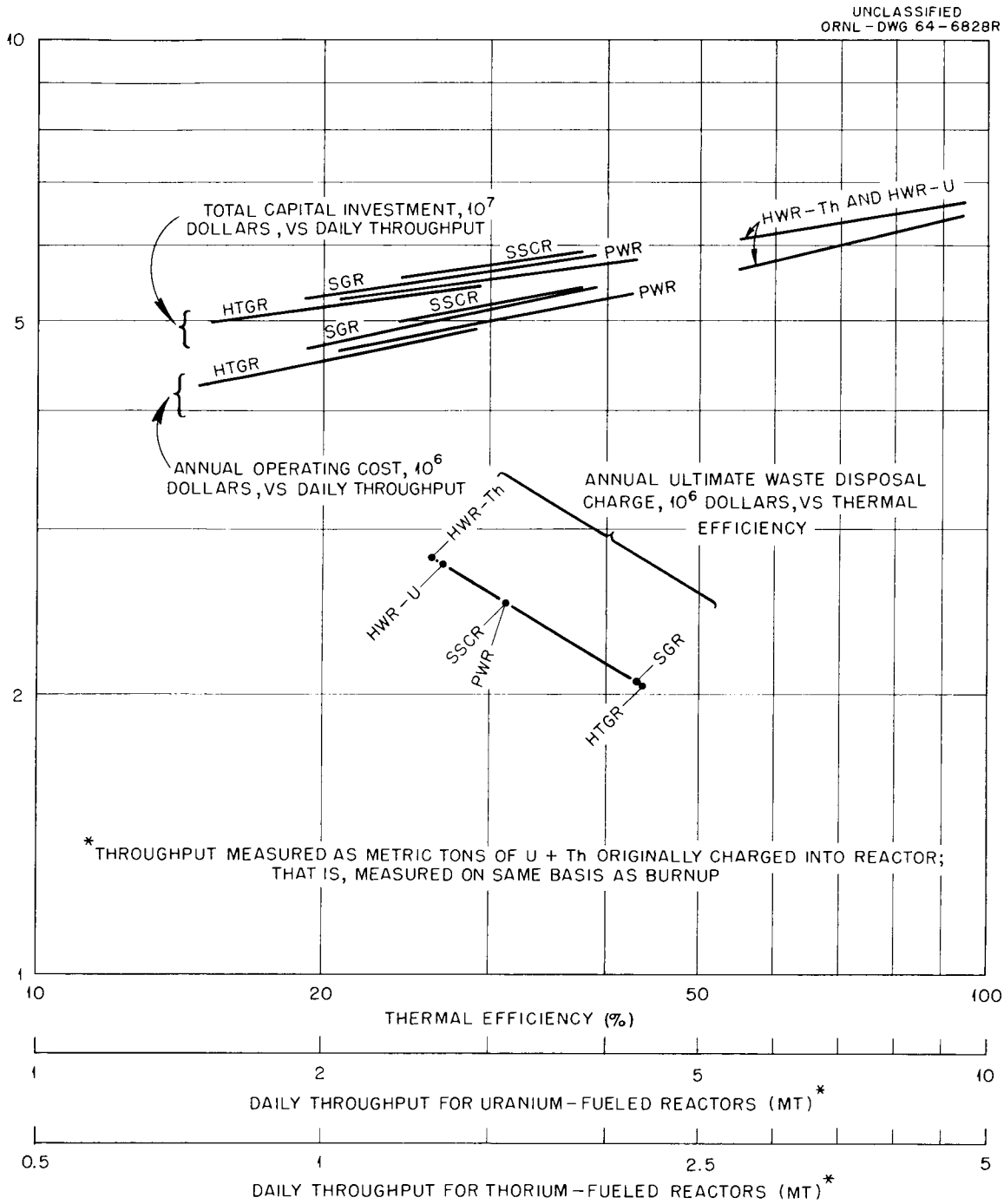

Fig. 9.3. Capital, Operating, and Ultimate Waste Disposal Costs for a Single-Purpose Reprocessing Plant Serving a 15,000-Mw(e) Nuclear Power Industry. Based on standard two-to-eight day turnaround formula.

Table 9.1. Spent-Fuel Processing Cost Estimates for Reference Single-Purpose Plants[a]

Basis: 15,000-Mw(e) total capacity, 80% load factor, 1000 Mw(e) per reactor; 310 operating days per year (processing plus turnaround time); HTGR case is for separate recovery of uranium particles for resale

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Reference material burnup, Mwd/MT	21,248	29,580	12,540	39,600	48,500	16,750
Thermal efficiency, %	31.1	31.2	26.8	26.1	44.4	43.6
Throughput rates, MT/year						
Uranium	645	34.9	1280	13.9	7.14	587
Thorium		426		393	189	
Plutonium	4.11		5.93			3.42
Fission products	14.1	14.0	16.3	16.7	10.0	10.0
Reference material	663	475	1300	424	206	600
Batches processed per year	19.2	6.91	57.1	42.4	54.0	48.7
Processing days per year	156	255	155	155	155	155
Processing rates						
Reference material, MT/day	4.23	1.86	8.40	2.73	1.37	3.87
Plutonium or highly enriched uranium, kg/day	26.3	137	8.26	89.7	217	22.1
Fissile plutonium and uranium, kg/day	44.7	75.8	38.3	39.7	102	73.5
Total capital investment, 10^6 $	58.5	59.3	65.8	61.1	57.9	59.0
Annual operating cost, 10^6 $	5.37	5.47	6.33	5.69	5.30	5.45
Ultimate disposal charge, 10^6 $/year	2.52	2.51	2.75	2.79	2.05	2.05
Cost at 15% fixed charges						
Total, 10^6 $/year	16.7	16.9	19.0	17.6	16.0	16.4
Unit, $/MT of reference material	25,200	35,600	14,600	41,500	77,700	27,300
Unit, mills/kwhr(e)	0.159	0.161	0.181	0.167	0.152	0.156
Cost at 22% fixed charges						
Total, 10^6 $/year	20.8	21.0	23.6	21.9	20.1	20.5
Unit, $/MT of reference material	31,400	44,200	18,200	51,700	97,600	34,200
Unit, mills/kwhr(e)	0.198	0.200	0.224	0.208	0.191	0.195
Cost at 30% fixed charges						
Total, 10^6 $/year	25.4	25.8	28.8	26.8	24.7	25.2
Unit, $/MT of reference material	38,300	54,300	22,200	63,200	120,000	42,000
Unit, mills/kwhr(e)	0.242	0.245	0.274	0.255	0.235	0.240

[a]See Table 9.2 for alternate "Off-Reference" cases.

separated, but the uranium particles are discarded with the fission products, while the thorium particles are processed to recycle their contained thorium and uranium. The difference between the two estimates is the extra cost of recovering the uranium for resale. At 22% fixed charges and for the particular case shown, approximately 1020 kg/year of fissile uranium isotopes (less losses) is recovered at an extra cost of $2,100,000. Figures 9.1, 9.2, and 9.3 show results of calculations, not included in Tables 9.1 and 9.2 but reported in Appendix D, for the case of a single type of fuel particle with simultaneous recovery of all the thorium and

Table 9.2. Spent-Fuel Processing Cost Estimates for Off-Reference Single-Purpose Plants

Basis: 15,000-Mw(e) total capacity, 80% load factor, 1000 Mw(e) per reactor; 310 operating days per year (processing plus turnaround time); HTGR case is the same as in Table 9.1 except that uranium particles are assumed to be discarded as high-level radioactive waste; difference represents the extra cost of recovering the uranium particles separately

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Reference material burnup, Mwd/MT	32,343	42,531	17,540	23,000	48,500	33,750
Thermal efficiency, %	31.1	31.2	26.8	26.1	44.4	43.6
Throughput rates, MT/year						
Uranium	418	28.8	911	24.5	4.97	285
Thorium		287		688	189	
Plutonium	3.17		4.68			2.58
Fission products	14.1	14.0	16.3	16.7	10.0	10.0
Reference material	435	330	932	730	206	298
Batches processed per year	12.5	4.63	40.7	73.0	27.0	23.7
Processing days per year	210	273	155	155	155	155
Processing rates						
Reference material, MT/day	2.08	1.21	6.01	4.71	1.28	1.92
Plutonium or highly enriched uranium, kg/day	15.1	106	30.2	158	32.1	16.6
Fissile plutonium and uranium, kg/day	23.3	58.4	34.0	83.3	26.6	49.2
Total capital investment, 10^6 $	52.7	55.6	62.1	67.5	51.8	52.9
Annual operating cost, 10^6 $	4.63	5.00	5.83	6.55	4.53	4.69
Ultimate disposal charge, 10^6 $/year	2.52	2.51	2.75	2.79	2.05	2.05
Cost at 15% fixed charges						
Total, 10^6 $/year	15.1	15.9	17.9	19.5	14.4	14.7
Unit, $/MT of reference material	34,700	48,200	19,200	26,700	69,900	49,300
Unit, mills/kwhr(e)	0.144	0.151	0.170	0.185	0.137	0.140
Cost at 22% fixed charges						
Total, 10^6 $/year	18.7	19.7	22.2	24.2	18.0	18.4
Unit, $/MT of reference material	43,000	59,700	23,800	33,200	87,400	61,700
Unit, mills/kwhr(e)	0.178	0.187	0.211	0.230	0.171	0.175
Cost at 30% fixed charges						
Total, 10^6 $/year	23.0	24.2	27.2	29.6	22.1	22.6
Unit, $/MT of reference material	52,900	73,300	29,200	40,500	107,300	75,800
Unit, mills/kwhr(e)	0.219	0.230	0.259	0.282	0.210	0.215

uranium for recycle. This cost is almost the same as for the discard case, so the "extra" uranium is recovered almost "free," but the uranium isotopes are all mixed together rather than being segregated and hence the physics advantages of the two-particle scheme are not available.

Processing costs in dual-purpose plants for servicing units of 20,000-Mw(e) capacity were calculated based on the original "guesstimates" of the optimum burnups and the expected thermal efficiencies,

batch sizes, fuel compositions, etc., and time did not permit recalculating these after the physics and other studies were done, as was partly possible for the single-purpose cases presented in Tables 9.1 and 9.2 and Figs. 9.1, 9.2, and 9.3. Thus the single- and dual-purpose plant costs presented in Table 9.3 are not directly comparable with the single-purpose costs presented in the other tables and figures. The numbers in Table 9.3 do, however, give a fair comparison between single-purpose plants and dual-purpose plants and between the various fuel types with respect to how well they do or do not "mix" with PWR fuel in a dual-purpose plant. The PWR-PWR case in Table 9.3 shows that increasing the size of a single-purpose PWR plant by 33% permits a 20% reduction in the unit processing cost. The other cases in Table 9.3 show that, on the average, going from single-purpose plants to dual-purpose plants permits unit cost savings in every case, but not as much as 20% in any case because of the extra costs associated with plants that must handle two different types of fuel instead of only one. In these examples the two other uranium fuels (HWR-U and SGR fuel) combined with PWR fuel to give almost the same cost advantage, percentagewise, as the PWR-PWR case; the two ThO_2 fuels (SSCR and HWR-Th fuel) mixed with the PWR fuel with somewhat less advantage; and the HTGR fuel mixed least well, giving a cost saving of only 12% in this particular case.

This comparison of single- and dual-purpose plants for servicing, respectively, units of 15,000- and 20,000-Mw(e) capacity is based on the "average of PWR and second fuel processing costs," as indicated in the table, rather than on a division of costs between two fuel types in the dual-purpose plants. The AEC-NFS "revenue day" method of assigning costs in proportion to the number of processing-plus-turnaround days for each fuel is simple to apply and could have been used as a basis for this comparison, but it is not considered to be a fair method for the evaluation of differences that exist between reactor types, since it does not reflect the actual manner in which two fuel types contribute to total costs in a plant designed to fit the combined fuel load in each case. It has the further disadvantage, for this study at least, of giving a different PWR cost for each dual-purpose plant, covering a range from 0.106 to 0.206 mills/kwhr(e), and thus complicating the comparison between the various other fuel types. It would seem that for a comparative evaluation the relative ranking of processing costs for various fuels after dividing the dual-purpose plant costs between them should be the same as that obtained on the basis of total annual dual-purpose plant costs, since the cheapest total should be the best for the overall economy. This implies that the same PWR cost should be assigned in each case because this is the only sure method of always obtaining the same relative ranking before and after division of costs between fuel types. The lowest PWR cost that can be justified on this basis is its cost in the 20,000-Mw(e) PWR (single-purpose) processing plant, since this is equivalent to assigning all the extra costs of the dual-purpose plant to the non-PWR type fuel. The last section of Table 9.3 lists costs assigned to each fuel type by three different methods. The "revenue day method" is seen to assign to the HWR fuels a <u>higher</u> cost than they had in their single-purpose plants, thus giving PWR <u>more than</u> all the advantage of the dual-purpose plant economies. The "constant PWR" example shows the highest

Table 9.3. Cost Savings in Dual-Purpose Processing Plants Compared with Single-Purpose Plants

Basis: 22% fixed-charge rate; 1000 Mw(e) per reactor; 802 load factor; single-purpose plants service 15 reactors of the same type; dual-purpose plants service 10 PWR reactors plus 10 reactors of a second type, as indicated; the PWR-PWR "dual-purpose" plant is actually a single-purpose plant for the purpose of comparing the economics of scale at the 15,000- to 20,000-Mw(e) level when there is no added complication of a second fuel type

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Burnup, Mwd/MT	25,000	30,000	15,400	22,000	51,400	35,000
Thermal efficiency, %	31.1	31.2	26.8	26.8	45.8	43.6
Processing cost in single-purpose plant, mills/kwhr(e)	0.198	0.195	0.221	0.226	0.170	0.178
Average of PWR and second fuel processing costs in single-purpose plants, mills/kwhr(e)	0.198	0.196	0.210	0.212	0.184	0.188
Average of PWR and second fuel processing costs in dual-purpose plant, mills/kwhr(e)	0.158	0.164	0.170	0.177	0.162	0.153
Average savings in processing costs for the two fuel types, %	20.2	16.3	19.0	16.5	12.0	18.6
Cost assigned to second fuel, mills/kwhr(e)						
"Revenue Day"	0.158	0.160	0.222	0.248	0.128	0.115
"Constant PWR"	0.158	0.170	0.182	0.195	0.166	0.147
"Average Savings"	0.158	0.163	0.179	0.189	0.150	0.145

costs that should be assigned to the non-PWR fuel. The "average savings" example gives slightly lower costs to the non-PWR fuel by sharing the extra dual-purpose plant costs between both fuel types. In the absence of detailed calculations, it is suggested that the average percentage savings shown in Table 9.3 be applied to the single-purpose cost estimates of Tables 9.1 and 9.2 and Figs. 9.1, 9.2, and 9.3 to generate dual-purpose plant estimates for those cases.

Summary

Costs were estimated for processing irradiated fuel from the seven reactor types under consideration, over a range of burnup values, in single- and dual-purpose processing plants sized to exactly match the fuel load indicated. Each fuel type was considered separately for the case of processing in a single-purpose plant designed to service 15,000-Mw(e) power plants of the given reactor type. Alternatively, the costs for dual-purpose plants serving 10,000-Mw(e) plants using pressurized-water reactors plus 10,000-Mw(e) plants using a given reactor type were also estimated. Capital and operating costs were calculated for each

individual case, with annual fixed charges on capital investment assumed to be 22% to reflect private investor ownership, including depreciation, interest on debt capital, return on equity capital, taxes, and insurance. Alternative fixed-charge rates of 15 and 30% were also evaluated.

The estimates are comparable with those previously made for a large nuclear water-desalination industry[5-9] but are higher, in general, primarily because of the higher fixed charge rate and, secondarily, because of other differences. The calculated costs were based on the estimates of Farrow,[4] with increases to allow for escalation to 1964–1965 prices and to add radioactive off-gas treatment facilities, land, startup, working capital, and ultimate waste disposal costs. Modifications were made as appropriate in the processing of each fuel type to allow for differences in turnaround time, in fertile- and fissile-material throughput rates, and in head-end treatments required.

Nominal plant sizes covered a range of approximately 1 to 10 metric tons per day, with plutonium or enriched uranium throughputs from 15 to 150 kg/day and fission-product throughputs from 10 to 20 metric tons/yr. Because of the relatively low cost-scaling factors typical of radiochemical processing plants, however, the costs varied over a much narrower range than the processing rates. Total capital investments varied from $\$50 \times 10^6$ to $\$70 \times 10^6$, annual operating costs from $\$4 \times 10^6$ to $\$6 \times 10^6$, and annual ultimate waste disposal charges from $\$2 \times 10^6$ to $\$3 \times 10^6$. Expressed in mills/kwhr, for single-purpose plants at a 22% fixed-charge rate, the calculated range of costs was 0.15 to 0.25, with the high-burnup, high-efficiency reactors at the low end, as might be expected from a comparison of the relative amounts of fertile, fissile, and fission-product material to be processed to service the same size electrical economy.

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10. FUEL SHIPPING COSTS

R. Salmon*

The fuel cycles include fuel shipments of the following types:

1. Irradiated (spent) fuel of high gamma activity shipped from reactor to reprocessing plant,
2. Recycled fresh fuel of moderate gamma activity shipped from fabrication plant to reactor,
3. Nonrecycled fresh fuel of zero gamma activity shipped from fabrication plant to reactor.

The shipping cost was calculated in dollars per kilogram of uranium or thorium charged to the reactor and also in mills per kilowatt-hour of electricity generated.

Shipping cost is the total of the following four items:

1. The handling cost, which covers the cost of loading, unloading, rigging, testing, and decontamination,
2. The insurance cost, which covers insurance against loss of or damage to the fuel in transit,
3. The freight cost, which covers the cost of transportation charged by the railroad,
4. The cask cost, which covers the annual fixed charges on the shipping casks (recovery of investment, return on investment, taxes, and maintenance).

The results of the shipping cost calculations are summarized in the following tables:

Table 10.1. Spent Fuel Shipping Costs as a Function of Burnup
Table 10.2. Spent Fuel Shipping Cost Breakdown
Table 10.3. Recycled Fresh Fuel Shipping Costs
Table 10.4. Nonrecycled Fresh Fuel Shipping Costs

Spent Fuel Shipping Costs

The irradiated fuel elements removed from the reactors, being highly gamma-active, require heavily shielded casks. Shipping costs were calculated by means of a computer code designated NORA, which calculated the dimensions of the casks and then the associated costs. NORA is similar to an earlier code, MYRA,[1] but includes certain additional optimizing features. For each reactor, the shipping cost was determined at several burnup levels. The cost basis and assumptions used are given below:

1. Shipment is by rail. Distance is 1000 miles each way.
2. Compliance with ICC and AEC regulations is required.
3. Round-trip time is 16 days.

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*This work was done with the advice and assistance of J. T. Roberts.

Table 10.1. Spent Fuel Shipping Costs As a Function of Burnup

Reactor	Burnup (Mwd/MT of U or Th)	Minimum Feasible Cooling Time (days)	Actual Cooling Time (days)	Delivery Time (days)	Total Time[a] (days)	Shipping Cost ($/kg of U or Th)	Shipping Cost [mills/kwhr(e)]
PWR	20,000	120	150	8	197	2.41	0.0161
	25,000	150	150	8	201	2.92	0.0157
	30,000	150	150	8	206	3.39	0.0151
	35,000	150	150	8	210	4.21	0.0161
SSCR	20,000	<60	90	24	168	5.27	0.0353
	25,000	<60	90	24	171	5.55	0.0297
	30,000	<60	90	40	195	5.80	0.0258
	35,000	<60	90	40	202	6.07	0.0232
	40,000	<60	90	40	210	6.42	0.0215
HWR-U	14,000	120	240	8	281	1.62	0.0180
	15,400	120	240	8	282	1.66	0.0168
	17,000	120	240	8	282	1.71	0.0156
	20,000	120	270	8	313	1.73	0.0127
HWR-Th	15,000	120	120	43[b]	195	4.18	0.0433
	22,000	120	120	42[b]	194	5.07	0.0359
	30,000	120	150	43[b]	225	4.57	0.0237
	40,000	120	150	54[b]	236	5.25	0.0204
HTGR	40,000	<60	120	8	166	18.50	0.0421
	51,400	<60	150	8	201	19.00	0.0337
	60,000	<60	150	24	220	18.46	0.0280
	70,000	<60	150	24	224	18.52	0.0241
HTGR[c]	40,000	<60	120	8	166	6.00	0.0137
	51,400	<60	120	8	171	6.04	0.0107
	60,000	<60	120	8	174	6.21	0.0095
	70,000	<60	120	8	178	6.39	0.0083
SGR	15,000	<60	120	8	161	2.35	0.0150
	25,000	<60	120	8	193	2.58	0.0099
	35,000	<60	120	8	165	2.87	0.0078
	45,000	<60	120	8	168	3.45	0.0073

[a]Total time from discharge of element from reactor to completion of processing.

[b]Measured from time of discharge of first element from cooling area.

[c]Fuel particles separated and canned separately. Cost shown includes cost of shipping both canned particles and element bodies.

4. Casks are empty on return trip.
5. Casks are shared among 15 identical reactors.
6. Handling cost is $500 per round trip.
7. Insurance charge is 0.004 times the value of the fuel shipped.
8. Freight cost is $0.0193 per lb for the loaded cask and $0.0181 per lb for the empty cask.
9. Maximum weight of cask plus fuel is 220,000 lb.
10. Casks are purchased. Cask cost is $1.00 per lb of cask weight. Fixed charge rates on casks are 0.15 per year for nonshared casks and 0.22 per year for shared casks. When casks are shared, utilization time is limited to not more than 80%.
11. Criticality calculations were not made. It was assumed that criticality could be avoided by using boron-containing divider plates.

Table 10.2. Spent Fuel Shipping Cost Breakdown

Reactor	Burnup (Mwd/MT of U or Th)	Elements per Cask	U or Th per Cask (kg)	Number of Casks	Shipping Costs ($/kg of U or Th)[a]				
					Handling	Insurance	Freight	Casks	Total
PWR	25,000	10	4398	9	0.12	0.51	1.64	0.65	2.92
SSCR[b]	30,000	4	1700	16	0.29	1.38	3.09	1.04	5.80
HWR-U	15,400	43	5749	11	0.09	0.09	1.10	0.38	1.66
HWR-Th	22,000	16	1878	2[c]	0.27	0.87	3.04	0.89	5.07
HTGR	51,400	46	649	20	0.78	1.59	11.98	4.65	19.00
HTGR[d]	51,400	30	3225	4	0.16	1.59	2.37	0.92	6.04[e]
SGR	35,000	33	4267	4	0.12	0.57	1.64	0.54	2.87

[a]Costs for PWR, HWR-U, and SGR are in dollars per kilogram of uranium charged to reactor. Costs for SSCR, HWR-Th, and HTGR are in dollars per kilogram of thorium charged to reactor.

[b]Fuel element disassembled into four sections.

[c]Not shared.

[d]Fuel particles separated and canned separately. Cost shown includes cost of shipping both canned particles and element bodies.

[e]$1.00 canning charge added.

Table 10.3. Recycled Fresh Fuel Shipping Costs

Reactor	Burnup (Mwd/MT of U or Th)	Delivery Time (days)	Insurance Cost ($/kg of U or Th)	Shipping Cost[a] ($/kg of U or Th)[b]
PWR	25,000	8	1.13	1.49
SSCR	30,000	8	1.74	2.35
HWR-U	15,400	8	0.26	0.52
HWR-Th	22,000	8	1.00	1.70
HTGR	51,400	8	1.71	6.65
HTGR[c]	51,400	8	1.71	5.05
SGR	35,000	8	1.42	1.90

[a]Includes insurance cost.

[b]Costs for PWR, HWR-U, and SGR are in dollars per kilogram of uranium charged to reactor. Costs for SSCR, HWR-Th, and HTGR are in dollars per kilogram of thorium charged to reactor.

[c]Fuel particles separated and canned separately. Cost shown includes cost of shipping both canned particles and element bodies.

Table 10.4. Nonrecycled Fresh Fuel Shipping Costs

	Weight of One Element (lb)	Reference Material per Element (kg)	Insurance Cost ($/kg of reference material)	Shipping Cost[a] ($/kg of reference material)[b]
PWR	1,600	439.8	1.13	1.49
SSCR	1,655	425	1.74	2.13
HWR-U	353	133.7	0.26	0.52
HWR-Th	353	117.4	1.00	1.30
HTGR	280	14.1	1.71	3.70
SGR	618	129.3	1.42	1.90

[a]Includes insurance cost.

[b]Costs for PWR, HWR-U, and SGR are in dollars per kilogram of uranium charged to reactor. Costs for SSCR, HWR-Th, and HTGR are in dollars per kilogram of thorium charged to reactor.

12. Individual canning of fuel elements is not required.

13. Maximum allowable temperature of fuel element cladding is 950°F.

14. Decay products and fission-product activity were calculated as though the entire reactor power came from U^{235} fissions.

15. For the calculation of inventory charges, the spent fuel value was based on its fissile content plus ThO_2 at $5 per lb and minus cost of reprocessing.

16. For all reactors, except HWR-Th, the reprocessing batch size is equal to the discharge batch size. For HWR-Th, various reprocessing batch sizes were tried, ranging from 50 to 200 elements.

17. All elements in a cask were assumed to have the same exposure. This exposure is the average exposure multiplied by the peaking factor.

Description of Cask

The cask is a right circular cylinder with the opening at one end. The outer shell is 1.25-in.-thick steel and has cooling fins. The inner shell is 0.50-in.-thick steel. Between the shells is approximately 8 in. of lead shielding. The space inside the inner shell is divided into a gridwork of square pigeonholes by means of copper-1% boron divider plates. One fuel element (assembly) fits into each pigeonhole with 1/4-in. clearance all around. The fuel elements are cushioned and restrained by metal spring clips. The lid of the cask is gasketed and bolted in place.

Heat removal from this cask is by natural convection and radiation. There is no mechanical cooling system. The cask is air-filled.

The weight of the loaded cask is limited to 220,000 lb, based on carrying capacity of standard railroad cars and handling equipment. An allowance of 20,000 lb was made for tie-downs and other appurtenances, making a total load of 240,000 lb.

Compliance with Shipping Regulations

Shipments must be made in compliance with existing ICC[2] and proposed AEC[3] regulations. Based on these regulations, the following specific restrictions were placed on the cask design:

1. Dose Rate. The dose rate shall not exceed 10 mr/hr at a distance of 100 cm from the outermost fuel element.

2. Heat Removal. Decay heat shall be removed without allowing the metal cladding temperature of any part of the fuel element to exceed 950°F.

3. Criticality. No restriction was set on the assumption that criticality can be adequately avoided by using boron in the divider plates.

4. Fire Protection. Void spaces are provided so that if the lead shielding is melted, its expansion will not burst the cask.

Method of Calculation

A block flow chart of NORA is shown in Fig. 10.1. Besides calculating the cask dimensions and shipping costs, NORA goes through certain optimization procedures to find conditions that give minimum total cost. The

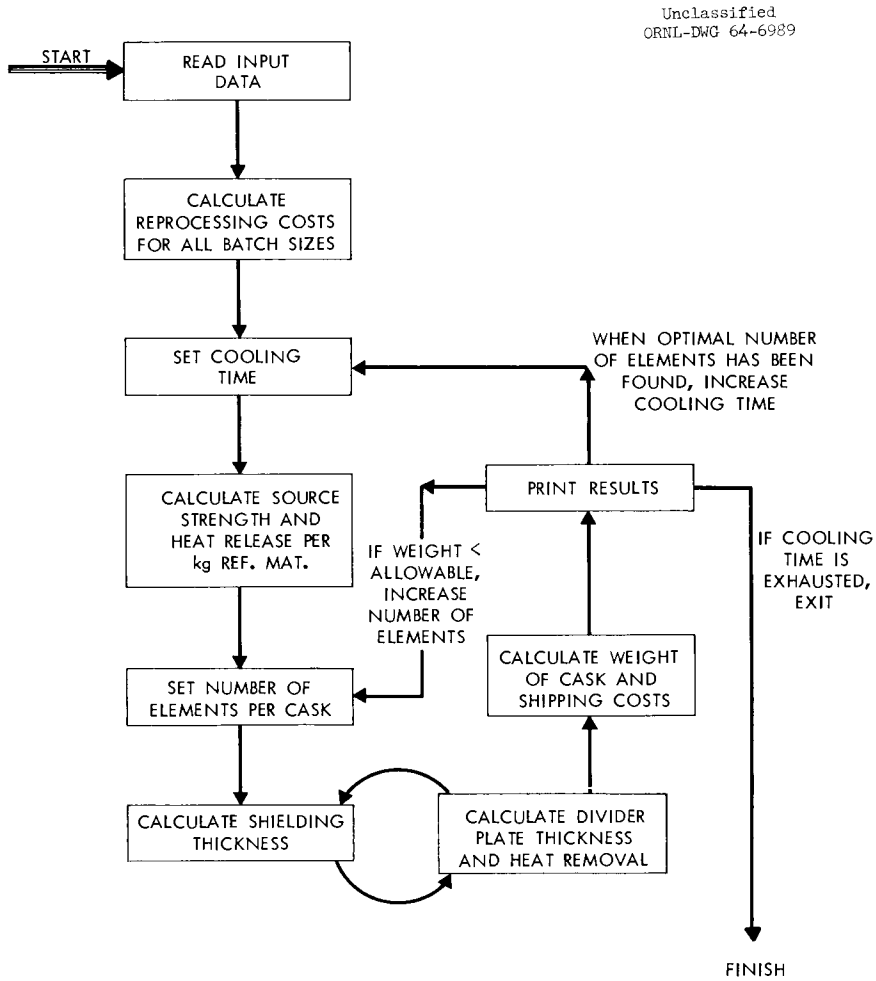

Fig. 10.1. Block Flow Diagram of NORA Code.

conditions sought are the optimal cooling time, the optimal number of elements per cask, the optimal number of casks, and the optimal number of fuel elements per reprocessing batch.

Input Data. The required input data include

1. Fuel dimensions, weights, value, and other pertinent characteristics,
2. Burnup and specific power,
3. Cooling time (minimum, maximum, and difference),
4. Number of reactors,
5. Cost factors associated with shipping,
6. Cost factors associated with reprocessing,
7. Inventory charge rate and other economic factors,
8. Elements per reprocessing batch (minimum, maximum, and difference),
9. Material properties,
10. Maximum allowable temperature of fuel cladding.

The input data used for each of the reactors studied are given in Table 10.5.

Description of Code. Assuming all the input data are received, NORA proceeds as follows:

1. Reprocessing costs are calculated for each batch size.
2. The cooling time is set at the minimum specified.
3. Using the burnup, specific power, peaking factor, and cooling time, the gamma source strength and decay heat release are calculated per fuel element. The subroutine used here is the PHOEBE code of Arnold, which is based on the data of Blomeke and Todd[4] and of Knabe and Putnam.[5]
4. The number of elements per cask is set at one.
5. The shielding thickness is calculated by trial and error until the dose is between 90 and 100% of the allowable dose.
6. The thickness of the copper-boron heat-conducting divider plates is calculated by trial and error until the required rate of heat removal is achieved without exceeding the allowable temperature.
7. When the shielding and heat-removal relationships have been simultaneously satisfied, the cask design is complete. The cask weight and all costs are calculated. The optimal number of casks is found. The results are printed out.
8. The number of elements per cask is doubled, and steps 5, 6, and 7 are repeated.
9. When the number of elements per cask becomes so large that the weight limit is exceeded, an interpolation procedure takes over, and the calculation converges toward the maximum number of elements that can be carried within the weight limit. The optimal number of elements per cask is then searched for and reported.
10. The cooling time is increased by the specified amount, and steps 3 through 9 are repeated.
11. When the maximum cooling time is determined, the calculation ends.

Optimization Procedures

The shipping costs reported are for the optimal conditions of shipment for each reactor. That is, the cooling time, number of elements per cask, number of casks, and number of elements per process batch were chosen to minimize the total shipping, inventory, and reprocessing costs in each case. For a given reactor, changing the burnup changes the optimal conditions.

Inventory Charges. Inventory charges are based on the value of the fuel, the time it is held in storage, and the inventory charge rate. The storage periods of interest are

1. The cooling period,
2. The delivery period, which follows the cooling period, and which is the time required to deliver a full process batch to the reprocessing plant,
3. The preprocessing lag period arbitrarily introduced by the processor; assumed to be 30 days in each case.

Table 10.5. Input Data for Spent Fuel Shipping Cost Calculations

	PWR	SSCR	HWR-U	HWR-Th	HTGR	HTGR (canned particles)[a]	SGR
Reference material	U	Th	U	Th	Th	Th	U
Reactor power, Mw(th)	3220	3266	3747	3747	2270	2270	2350
Specific power, Mw/MT of reference material	27.74	49.2	54.3	46.4	29.3	29.3	30.6
Cooling time, minimum, days	60	60	60	60	60	60	60
Cooling time, maximum, days	240	300	360	360	180	180	300
Element length, overall, ft	12.92	13.33	16.5	16.5	20.0	16.0	17.5
Element length, active, ft	11.0	10.92	15.0	15.0	15.5	15.5	14.0
Element weight, lb/ft	123.7	124.1	21.4	21.4	14.0	26.7	35.3
Element surface area, ft^2	264.7	285	15.04	15.04	19.05	24.3	38.9
Reference material per element, lb	439.8	425	133.7	117.4	14.1	107.5	129.3
Pins per element	234	238	1	1	1	1	18
Pin outside diameter, in.	0.403	0.420	3.83	3.83	4.7	6.0	0.590
Pin spacing, center to center, in.	0.525	0.541	0	0	0	0	0.700
Thermal conductivity of copper plate, Btu/ft·hr·°F	210	210	210	210	210	210	210
Box size, inside, in.	8.86	9.11	4.33	4.33	5.2	6.5	4.75
Elements per process batch	88	156	172	50–200	915	120	99
Peaking factor	1.2	1.5	1.13	1.13	1.5	1.5	1.48
Pa in spent fuel, g/kg of Th	0	1.08	0	1.70	1.55	1.55	0
Allowable cladding temperature, °R	1410	1410	1060[b]	1060[b]	1410	960	1410
Thermal efficiency	0.311	0.312	0.268	0.268	0.458	0.458	0.436
Burnup, minimum, Mwd/MT of reference material	20,000	20,000	14,000	15,000	40,000	40,000	25,000
Burnup, maximum, Mwd/MT of reference material	35,000	40,000	20,000	30,000	70,000	70,000	45,000
Fuel value, $/kg of reference material	110–160[c]	345	23	217	398	398	74–264[c]

[a]Fuel particles separated and canned separately in 6-in. cans.

[b]At exterior surface of element.

[c]Fuel value depends on burnup.

An increase in the cooling time tends to reduce shipping costs, since less shielding is required and the heat-removal problem is reduced. On the other hand, after lengthy cooling the incremental gain is small and tends to be outweighed by the increase in cooling cost (inventory charges for the cooling period). The existence of an optimal cooling time is therefore indicated.

The delivery time can be reduced by using more casks; again, the gains diminish as the number of casks is increased. Thus an optimal number of casks is to be expected.

The delivery time can also be reduced by reducing the number of elements per process batch. This tends, however, to increase reprocessing costs, because small batches are penalized by higher turnaround costs. An optimal process batch size may therefore be found. (In the actual calculations, this was done only for the HWR-Th reactor, which uses continuous discharge. For the other reactors, the optimal process batch size was assumed equal to one reactor discharge batch.) The preprocessing lag time, being assumed constant, does not enter into any optimization procedure.

It is evident that a change in the inventory charge rate or in the fuel value may result in the selection of different optimal conditions. Lower inventory rates and lower fuel values favor longer cooling times, longer delivery periods, smaller numbers of casks purchased, and larger process batch sizes.

Optimal Number of Elements per Cask. The ratio of fuel payload to cask weight tends to increase as the number of elements per cask increases. Also, the number of shipments per batch decreases. These factors tend to reduce the shipping cost. In most of the cases studied, the optimal number of elements was the maximum number that could be carried within the weight limitation. However, the cost curve may go through a minimum in some cases; this is largely due to the increasing difficulty of heat removal as the number of elements per cask becomes large.

Optimal Cooling Time. For a given irradiated fuel, there is some minimum cooling time below which it is impossible to ship even one fuel element in a cask of the type described. The limiting factor is heat removal; the decay heat cannot be removed at the required rate without exceeding the allowable cladding temperature. Increasing the cooling time above this minimum reduces the amount of shielding needed and may also make it possible to ship a greater number of elements per cask. Both factors tend to reduce the shipping cost. As mentioned earlier, there is usually some optimal cooling time at which the sum of shipping cost and cooling cost is minimized. This is shown in Fig. 10.2 for the HWR-U reactor. Similar curves could be drawn for the other reactors.

Optimal Number of Casks. If the number of casks purchased is just adequate to keep up with the reactor discharge schedule, the delivery of each batch will be completed just before the next batch is discharged. The delivery time will thus be about equal to the time between discharges. The delivery time can be shortened by buying more casks; if enough casks are bought, the entire delivery can be made in one shipment, which will take eight days (half of the round-trip time). NORA starts with the minimum number of casks and then checks to see whether savings can be effected by using more casks. It chooses the number of casks that gives the minimum sum of cask cost and delivery time cost.

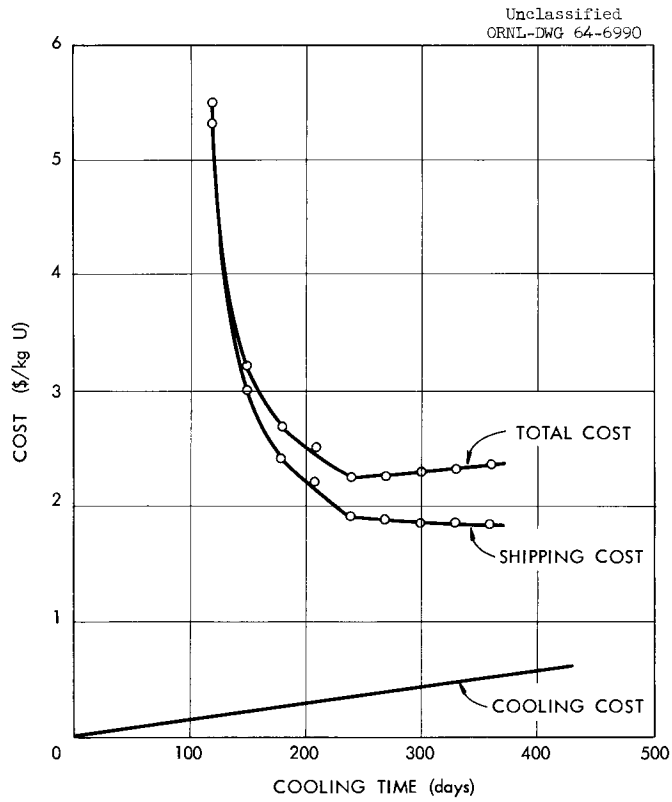

Fig. 10.2. Optimal Cooling Time for HWR-U Fuel Elements. Total cost minimized at a cooling time of 240 days; burnup, 15,400 Mwd/MT of U.

Cask Sharing. Since it would be wasteful for each of 15 identical reactors to have its own purchased casks, many of which would be idle a large part of the time, it was decided that similar reactors would share their casks. This gives a considerable saving in cask cost. Because of anticipated scheduling problems, it was decided to limit cask utilization to 80%. Fixed charges on casks were set at 22% per year on the assumption that centralized ownership of the casks could most conveniently be vested in the reprocessing plant. Where this gave higher costs than nonsharing, the nonshared basis was used.

Range of Economic Variables. The ground rules specified the following economic variables:

Annual inventory charge rate	6%, 10%, 12%
Annual fixed-charge rate for reprocessing plant	15%, 22%, 30%
Number of identical reactors	10, 15

The shipping costs shown in Table 10.1 are for the following conditions:

Annual inventory charge rate	10%
Annual fixed-charge rate for reprocessing plant	22%
Number of identical reactors	15

Calculations showed that the shipping costs for these conditions could be used for the entire range of the above variables without perceptible error. The first two do not enter into shipping cost directly, but do affect the optimal cooling time and optimal number of casks. It was found, however, that departures from the optimal cooling time of about 30 days did not appreciably affect total cost (shipping plus inventory charges). Similarly, small deviations from the optimal number of casks would not cause any significant error in total cost. The number of identical reactors enters into the cask cost because casks are shared. Again, however, the difference in sharing among 10 or among 15 reactors is negligible.

Sufficient calculations were made at 6 and 12% inventory rates, and at 15 and 30% fixed-charge rates on the reprocessing plant, to assure that the procedure described was satisfactory.

Recycled Fresh Fuel Shipping Costs

The reactors that use thorium fuel cycles will require shielded casks for shipment of recycled fresh fuel. These reactors are the SSCR, HWR-Th, and HTGR. The reactors that use uranium fuel cycles can ship their recycled fresh fuel in the same manner as their nonrecycled fresh fuel, that is, without shielding. For the uranium-fueled reactors, therefore, the shipping cost for recycled fresh fuel is the same as for nonrecycled fresh fuel. The assumptions used in the calculations for the thorium-based fuels are given below:

1. Assumptions 1 through 13 used for determining the spent fuel shipping costs are applicable.
2. The casks are not the same ones as those used for shipping spent fuel.
3. The type of cask used is the same as for the spent fuels except that only about 4 in. of lead is required. For the same gross weight, therefore, the number of fuel elements carried is considerably greater.
4. The number of casks purchased is such that delivery of a batch can be made in eight days.

The gamma activity of the recycled thorium fuels is due to the decay products of U^{232}, principally Bi^{212} and Tl^{208}. The U^{232} is produced in the reactor by a series of reactions initiated by an (n,2n) reaction on Th^{232}. The U^{232} decays to Th^{228}, with a half-life of 74 years; so both U^{232} and Th^{228} are present in the spent fuel. In chemical reprocessing of the irradiated fuel, the U^{232} comes out with the finished U^{233} product, and the Th^{228} comes out with the finished Th^{232} product. The subsequent members of the decay chain (those following Th^{228}) are eliminated with the fission products. Immediately after reprocessing, however, the decay products start to build up again, and both the U^{233} and the Th^{232} soon become gamma-active. This activity changes with time after reprocessing, so the amount of shielding required depends on how soon the shipment is made. Fabrication may require from 20 to 60 days. In addition, there may be a waiting period after fabrication, because it may take some time to accumulate a full cask load of fuel elements. Calculations showed, however, that the change of activity in the time range 30 to 100 days was small enough so that the effect on shipping cost was negligible. The

costs shown are based on a time lapse of 35 days between reprocessing and shipping.

For each fuel, the shipping cost was calculated in dollars per kilogram of reference material at only one burnup. The variation with burnup is negligible in the range being considered.

Nonrecycled Fresh Fuel Shipping Costs

Table 10.4 shows the calculated shipping costs for the fresh nonrecycled fuel elements. The assumptions and bases for the calculations are as follows:

1. No shielding is required.
2. Shipment is one-way only, a distance of 1000 miles.
3. The freight rate is $0.02 per lb.
4. The container weight is 25% of the fuel element weight.
5. The handling cost is $0.005 per lb of total load.
6. The insurance cost is 0.004 times the value of the fuel.
7. The container cost is $0.055 per lb of total load.

The result of these assumptions is a total shipping cost of $0.10 per lb of fuel element, plus insurance. Handling charges are based on the experience with natural uranium fuel elements described by Isakoff.[6]

Results

The last part of Table 10.2 shows a breakdown of the spent fuel shipping cost. The amount (kg) of uranium or thorium per cask is a good inverse indicator of shipping cost. The freight cost per cask load is about $7500 and handling is another $500, so these two important costs (in dollars per kilogram of uranium or thorium) vary inversely with the kilograms of uranium or thorium per cask. Elements that are large and bulky but contain relatively few kilograms of uranium or thorium (as HTGR) will have large costs.

The spent SSCR fuel elements were assumed to be disassembled prior to shipping. This was necessary in order to avoid excessively long cooling times. The HTGR has high burnup, but only 14 kg of Th per element and only one fuel pin per element, so it has no cooling problem. The reason the number of fuel pins per element is important is that the pins radiate heat to each other, so the centermost pin is the hottest.

In shipping the spent SSCR fuel elements, the 238-pin element was assumed to be divided into four sections. This shortened the minimum feasible cooling time from 270 days to less than 60 days. The reduction in total inventory time was equivalent to about $15 per kg of thorium at a burnup of 30,000 Mwd/MT of thorium. Further division of the fuel element did not yield any additional benefit.

Assuming that the SSCR elements can be conveniently disassembled, the large potential saving in inventory costs appears to make this operation economically justifiable. Facilities for disassembly of the elements will have to be provided at the reactors. The cost of disassembly has not been

included as part of the shipping cost, since it was felt that these costs could best be determined by those estimating the disassembly equipment requirements. In the fresh fuel shipping costs (Tables 10.3 and 10.4), the effect of the insurance cost is worth noting.

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11. PHYSICS ANALYSES OF ADVANCED CONVERTER REACTORS

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In performing reactor physics calculations for the reactors being evaluated in this study, primary emphasis was placed on determining the economic performance of these systems when operated with appropriate fuel management schemes, including fuel recycle in the thorium reactors. Therefore, the primary goal was the calculation of fuel feed requirements as a function of the discharge fuel burnup and discharge fuel compositions. Given these material balances over the history of reactor operation, the cost of power can be estimated for an assumed economic climate.

Since a number of reactor concepts were to be compared, efforts were made to perform physics calculations in a consistent manner for all the systems. It is difficult, however, to place physics calculations of two quite different concepts on a truly common basis. Calculational procedures proven for one concept may not yield consistent results for another. However, because of the time limitations imposed on this study, it was necessary to choose a calculational procedure that could be applied to all the concepts. Careful attention was given to details of cell and spectrum calculations in an effort to avoid prejudicing any concept by improper choice of physics techniques.

Zero-Dimensional Depletion Calculations

The zero-dimensional multigroup depletion code, TONG,[1] was used almost exclusively in this study. The point depletion model was chosen primarily to hold the computer time requirements to a reasonable level. Also, it appears to be a reasonable model to use for large reflected cores in which the power distribution is relatively uniform in space. In order to include the effects of the higher neutron leakage inherent with cores having a flattened power distribution, the core buckling was made larger than the geometric buckling. Where possible the buckling was normalized to leakage values calculated by the reactor sponsor. The TONG code was already in existence at ORNL, but extensive development work was required to allow description of the various fuel cycles proposed. A detailed description of the program is presented in Appendix E, and a few features are described here for completeness.

In the TONG calculation, a core that is refueled by the multibatch scatter-reloading technique can be treated by identifying the individual batches of fuel separately. Thus, calculation of a six-batch scatter-reloaded core requires following six sets of fuel nuclide concentrations during the depletion. The cycle sequence for removing the fuel batches can be specified, as well as the replacement fuel composition. After a suitable delay for processing (probably one or two refueling intervals), the recovered fuel may be returned to the core along with a specified

makeup fuel. Control can be maintained by various options on poison control, moderator composition changes, or changes in fuel compositions. A complete history of the operating life of a reactor can be established with one continuous computer calculation, with provisions for specifying the initial conditions at the start of each cycle.

Calculation of Spectra and Cross Sections

The TONG program uses modified versions of GAM-I[2] and THERMOS[3] to calculate fast and thermal group cross sections for use in depletion calculations. An extensive review of the basic cross sections was made at the start of this study.[4] Efforts were made to use the latest data from experiments by choosing the more reliable and consistent data.

In the case of U^{233}, there has been a troublesome uncertainty in the epithermal cross sections. A cross section set derived from various differential cross section measurements, mainly fast chopper measurements, yields an average capture-to-fission ratio (alpha) of about 0.23. Integral epicadmium measurements[5] have indicated that the average value of alpha in a $1/E$ spectrum above 0.5 ev is about 0.17.

The U^{233} cross sections used for most of this study were based on the differential measurements and, consequently, on the higher epithermal alpha value. However, very recent experiments at KAPL,[6] similar to those of Halperin, tend to confirm the lower value of alpha. A second cross-section set for U^{233} was generated by adjusting the epithermal cross sections of the first set, mainly in the energy range 1 to 1000 ev to produce resonance integrals and an average alpha in agreement with Halperin's integral measurements. Several of the depletion calculations were performed twice, using both cross section sets. It is recommended that results based on the low-alpha set be taken more seriously, and the difference in fuel-cycle performance resulting from this uncertainty in epithermal U^{233} cross sections is simply to be noted.

In the epithermal range, resonance parameters are used to determine shielded cross sections for all the fertile isotopes (including U^{234} and Pu^{240}) and for U^{236} and Pu^{242}. Cross sections for zirconium were also treated by resonance parameters to obtain shielded cross sections. Unshielded group cross-section data were used for all other nuclides, including the fissile fuels. The basic cross-section data are subject to continuous review by ORNL, and it is recognized that there will be differences with libraries used at other installations. Although some of the data used here may be changed in the future, at least the evaluations of all the concepts were based on the same data.

In order to treat spectrum changes with time, all point depletion calculations were made with not less than 11 fast groups (above 1.86 ev) and four thermal groups. In some cases, more groups were used in the fast or thermal range or in both ranges.

Optimization of Fuel Cycles

In the short time available, it was not possible to investigate in detail the behavior of each reactor concept. At the same time, it was desirable that the fuel cycle be optimized with respect to the specified economic parameters. This required defining an approach to equilibrium-cycle operation and optimizing the equilibrium cycle. A complete optimization of the startup cycle was not attempted, but efforts were made to prevent penalizing any concept by choice of a poor approach to equilibrium.

In order to optimize cycles for low fuel costs, the cycle times were varied through the device of specifying different values for the initial excess reactivity (with rods withdrawn) at the start of each cycle. The fuel enrichment or makeup required to provide the excess reactivity was determined by a direct iteration in an eigenvalue search. The reactor was kept critical during depletion by simulation of the specified shim control mechanism, such as uniform poison or moderator composition change.

In general, the procedure followed was to specify a value for the excess reactivity at the start of the first cycle and a single different value for the reactivity at the start of all the subsequent cycles. By this technique, the initial cycle was made longer than subsequent cycles. This was done in order that the first batch of fuel removed from a multi-batch core would not be discharged with an unreasonably low exposure. The choice of relative values of excess reactivity for initial and later cycles was made rather arbitrarily, but the first batch cycle was always made longer than the time between refuelings in the later cycles.

Fission-Product Treatment Used in These Studies

The time-dependent behavior of neutron absorptions in fission products is a major concern in any reactor depletion calculation. Therefore this problem was given careful attention in the calculations performed during this study.

Long Fission-Product Treatment

The basic approach involved a description of all the important fission-product nuclides in explicit isotope chains that were followed during the depletion calculation. This description treated 54 fission-product nuclides in 13 chains having a total of 22 connecting routes (shown in Fig. 11.1). The nuclides not treated in these explicit chains were lumped together as a single nonsaturating pseudoelement FP(1). This lumped fission product constitutes perhaps 5% of the total fission-product poisoning.

This description was dubbed the "Long Fission-Product Treatment — (LFP)" to distinguish it from a simpler treatment derived from it. The LFP treatment was taken as the standard against which simpler treatments were compared.

Treatment of the many fission products in detail reveals more loss than usually estimated with the traditional treatments using one of the

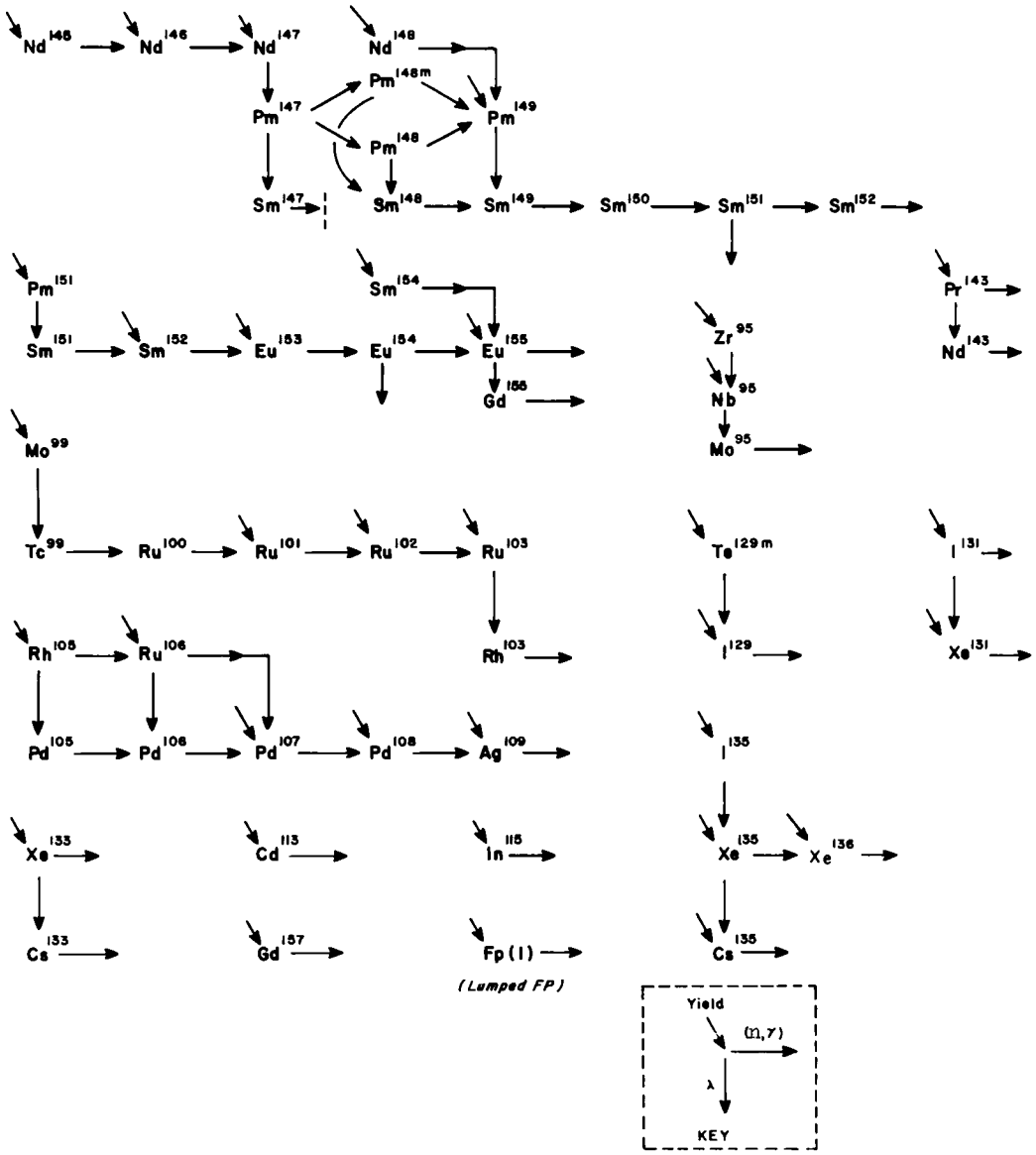

Fig. 11.1. Nuclide Chains Used in Long Fission-Product (LEP) Treatment in Depletion Calculations.

"lumped fission-product" sets. With the long burnups of interest to advanced converters, secondary fission-product routes become important and must be included.

Short Fission-Product Treatment

Since the computer time required for a depletion problem is dependent on the number of nuclide chains being treated, a simplified fission product description was used in many of the physics calculations. The nuclide chains for this treatment (called the Short Fission-Product Treatment, SFP) are shown in Fig. 11.2. It is obvious that many of the nuclide

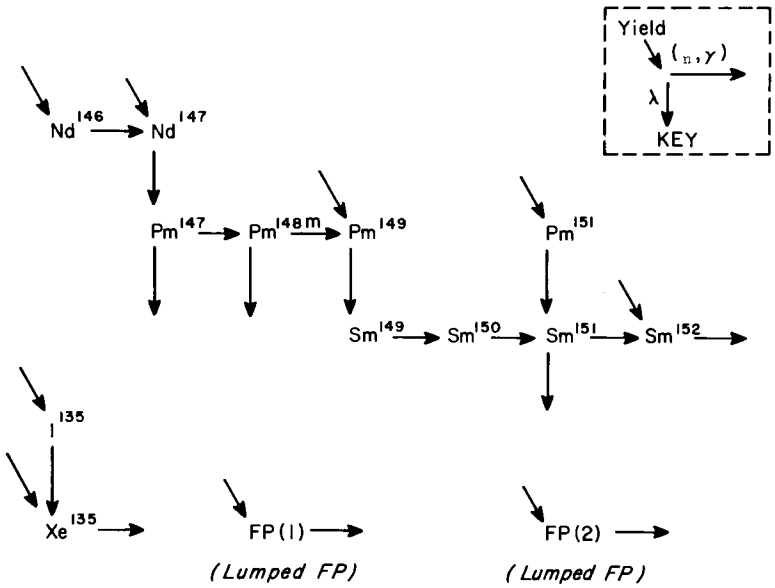

Fig. 11.2. Nuclide Chains Used in Short Fission-Product (SFP) Treatment in Depletion Calculations.

chains shown in Fig. 11.1 are not treated explicitly here. The nuclides involved in the omitted chains were lumped into a slowly saturating pseudoelement FP(2). The yields from fission and the cross sections used for both pseudoelements, FP(1) and FP(2), are presented in Table 11.1. The energy distribution of these lumped nuclide cross sections are estimated by weighting the point cross sections for the individual nuclides by their yield fractions. This simplified description was used in most of the optimization calculations, with the elaborate treatment being used to normalize the results.

Preliminary normalization calculations were performed for the PWR to determine the effective fission yield for FP(2) which would produce the same neutron losses to fission products as predicted by the LFP treatment.

Table 11.1. Fission Yields and Cross-Section Data for Lumped Pseudoelements

Fissioning Nuclide	FP(1)			FP(2)		
	Yield	σ_a (2200)	I_a	Yield	σ_a (2200)	I_a
U^{233}	1.41	3.86	7.0	0.88	33.7	173
U^{235}	1.31	2.74	5.7	1.02	34.2	185
Pu^{239}	1.06	2.74	5.7	1.4	49.1	210
Pu^{241}	1.06	2.74	5.7	1.4	49.1	210

The yields given for FP(2) in Table 11.1 are based on this normalization. As a further check the HTGR was analyzed with both treatments, and the calculated lifetimes were in good agreement.

Fertile and Fuel Nuclide Chains

The nuclide chains used to describe the time behavior of fuel and fertile nuclides are shown in Fig. 11.3. The chain associated with the (n,2n) reaction in thorium is of interest primarily in calculating the U^{232} content in spent fuel. This chain was omitted from most of the depletion calculations in this study.

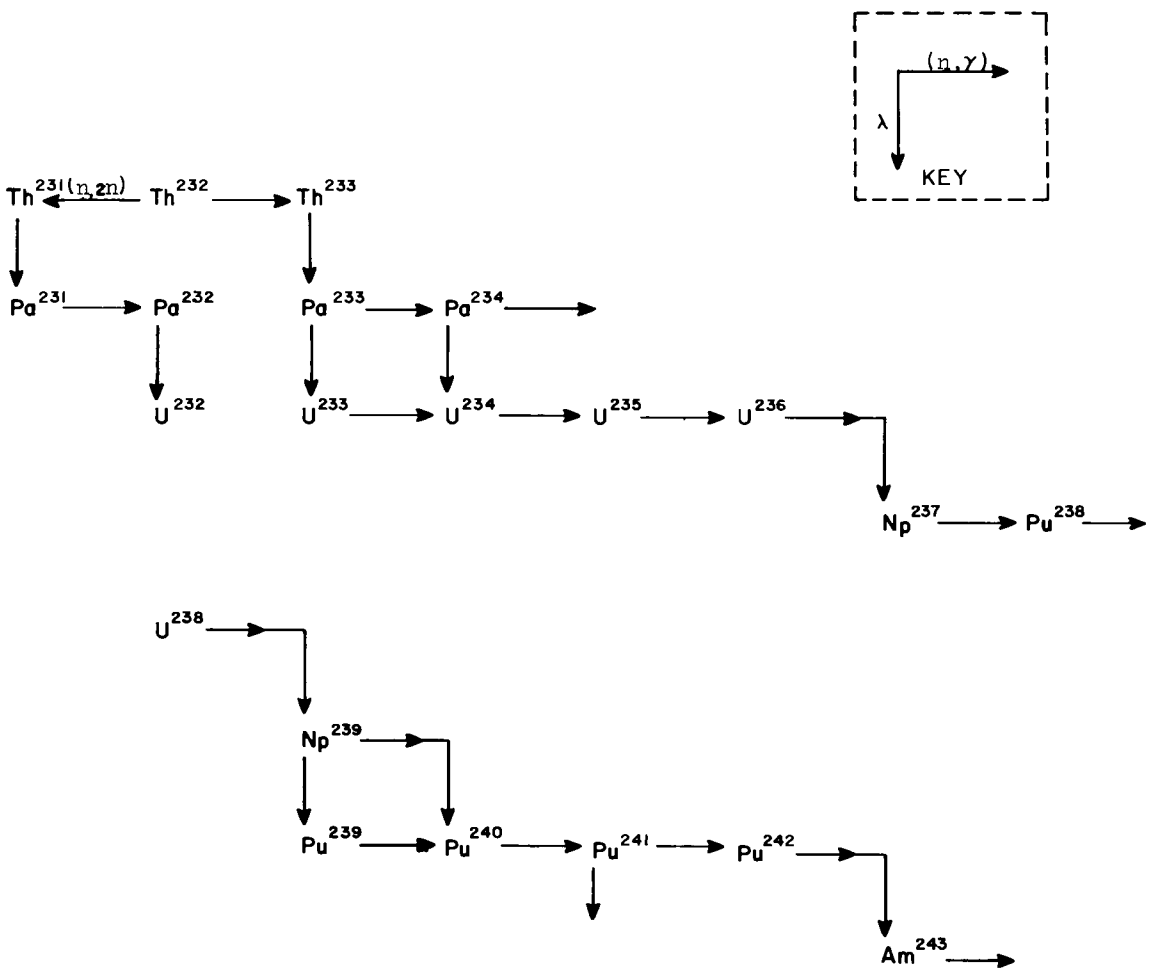

Fig. 11.3. Fuel Nuclide Chains Used in Depletion Calculations.

Physics of the 1000-Mw(e) PWR

Fuel-cycle calculations were performed for the Westinghouse reference design of a 1000-Mw(e) pressurized-water reactor. It was mutually agreed to substitute Zircaloy fuel cladding for the collapsed stainless steel cladding that was used in their reference design. The outside diameter of the clad rods was kept the same as in the reference design, and the cladding thickness was increased. Also, it was assumed that the fuel would be 92%-dense vibratorily compacted powder, rather than 96.5%-dense pellets. The combined effect of these two changes lowered the core loading of uranium from about 116 to about 103 MT.

Calculations were made with the TONG point depletion code, describing the neutron spectrum in 15 groups. The reference fuel management scheme was the "3-batch roundelay" or "scatter-reload" program in which one-third of the core loading is discharged at each shutdown. The reference design specified that the initial core loading have three batches of fuel (each occupying one-third the core) with enrichment ratios approximating the ratios existing at the start of an equilibrium cycle. This was simulated in the TONG calculations by specifying three batches of fuel (homogeneously mixed) having enrichment ratios (relative to average core enrichment) of 0.70, 1.0, and 1.30, respectively. The fuel with the lowest enrichment would be discharged at the first shutdown and replaced with fresh fuel. Excess reactivity (with rods withdrawn) at the start of each cycle was specified and the enrichment needed in the reloaded zone to obtain that reactivity was determined by a direct eigenvalue calculation. During the depletion cycle, a uniform poison concentration was determined to keep the reactor critical.

Initial excess reactivity (that is, feed fuel enrichment) was treated as a parameter in these calculations in order to optimize fuel costs. The feed and discharge enrichments for the equilibrium cycle are shown in Fig. 11.4 as a function of the average exposure of the discharged fuel. Also shown is the feed enrichment that would be required if the neutron leakage were 1% smaller (that is, a larger core size). These feed enrichments were compared with other published information and with results provided by Westinghouse. The results presented here predict enrichments (for a given fuel exposure) significantly lower than those presented by Westinghouse in their reference report for an alternative Zircaloy-clad core. For example, to achieve 24,000 Mwd/MT would require an equilibrium feed enrichment of about 2.4%, based on ORNL results, compared with about 3% indicated by Westinghouse. The ORNL results also predict a lower concentration of plutonium in the discharged fuel than does Westinghouse, which partially offsets the economic gain from the lower enrichment. Comparison of ORNL results with those of an MIT study[7] and a Combustion Engineering study[8] showed substantial agreement. The Westinghouse results have not been sufficiently detailed to indicate the source of the difference with their results. Based on the agreement with other studies, it appears that both the feed enrichment and final plutonium concentration should be lower than calculated by Westinghouse.

Detailed fuel-cycle performance data are presented in Tables 11.2 and 11.3. Neutron balances for case 7 (lowest fuel cost case) are presented in Table 11.4. Discharge plutonium concentrations are presented in Fig. 11.5.

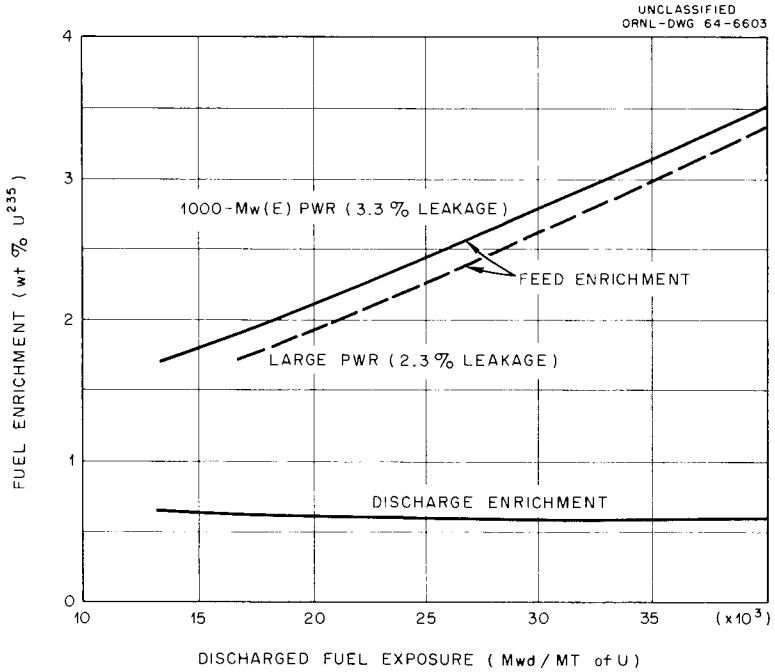

Fig. 11.4. PWR Feed and Discharge Enrichment Versus Exposure of Discharged Fuel.

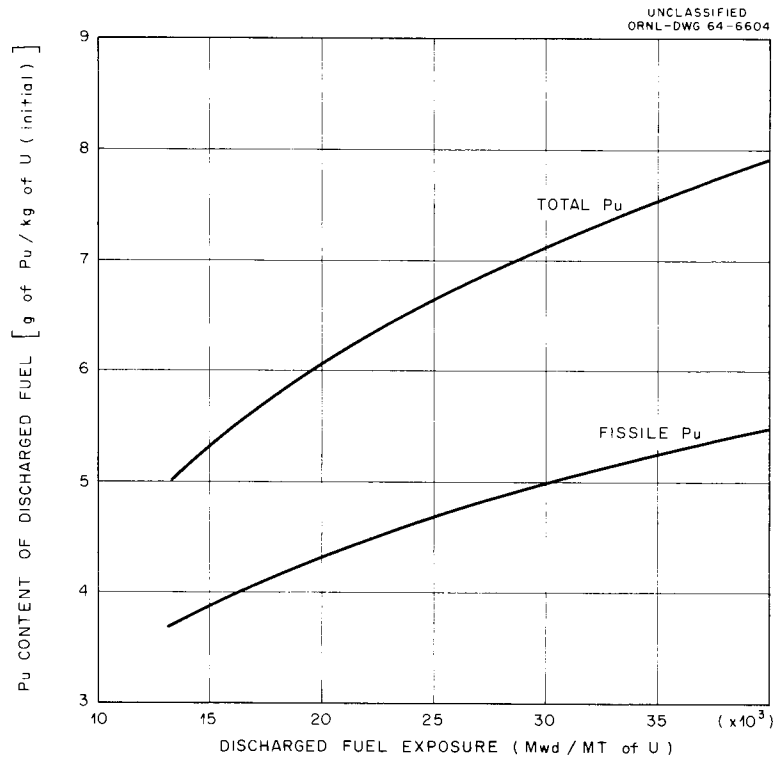

Fig. 11.5. Plutonium Concentrations in Fuel Discharged from PWR as a Function of Exposure of Discharged Fuel.

Table 11.2. Summary of Cycle Performance for PWR

Case	6	7	8	9	10	11
First cycle						
Initial k (no rod)		1.23	1.25	1.26	1.27	1.28
Cycle time, full-power days		378	425	458	485	513
Discharge exposure, Mwd/MT		11,975	13,475	14,552	15,397	16,290
Initial enrichment, wt % U^{235}		1.99	2.13	2.20	2.28	2.37
Equilibrium cycle						
Initial k (no rod)	1.10	1.13	1.15	1.16	1.17	1.18
Cycle time, full-power days	139	223	293	334	380	433
Discharge exposure, Mwd/MT	13,120	21,050	27,600	31,430	37,750	40,680
Feed enrichment, wt % U^{235}	1.70	2.20	2.66	2.94	3.27	3.66
Discharge enrichment, wt % U^{235}	0.651	0.630	0.592	0.592	0.597	0.608
Discharge Pu, g/kg of U feed	5.00	6.22	6.95	7.31	7.68	8.06
Pu composition, wt %						
Pu^{239}	62.19	56.03	53.52	52.56	51.76	51.11
Pu^{240}	23.68	23.74	22.92	22.39	21.79	21.15
Pu^{241}	11.43	14.93	16.57	17.07	17.59	18.04
Pu^{242}	2.74	5.30	7.09	7.98	8.85	9.69

Table 11.3. Mass Balance Over Life of PWR

Case	6	7	8	9	10	11
Equilibrium cycle initial k	1.10	1.13	1.15	1.16	1.17	1.18
History time (years)	30.30	30.44	30.63	30.24	30.35	31.42
Initial loading, kg						
U^{234}	18.5	20.5	22.0	22.8	23.6	24.5
U^{235}	1,866	2,060	2,206	2,286	2,371	2,461
U^{236}	9.4	10.3	11.1	11.5	11.9	12.4
U^{238}	101,456	101,456	101,456	101,456	101,456	101,456
Total feed material, kg						
U^{234}	365.6	309.5	290.2	279.9	276.4	282.9
U^{235}	36,713	31,078	29,142	28,107	27,757	28,414
U^{236}	184.3	156.0	146.3	141.1	139.4	142.7
U^{238}	2,096,800	1,386,600	1,082,200	946,930	845,470	777,830
Total material removed, kg						
U^{234}	266.9	198.2	168.9	154.9	145.1	140.4
U^{235}	13,883	8,731	6,850	6,085	5,567	5,286
U^{236}	4,132	3,994	3,949	3,882	3,892	4,035
U^{238}	2,074,400	1,365,200	1,061,600	927,110	826,050	758,230
Fissile Pu	7,847	6,170	5,300	4,853	4,530	4,362
Equilibrium cycle feed (1/3 core), kg						
U^{234}	5.8	7.6	9.2	10.2	11.4	12.8
U^{235}	583.0	760.3	922.8	1,024	1,143	1,286
U^{236}	3.0	3.8	4.6	5.1	5.7	6.5
U^{238}	33,819	33,819	33,819	33,819	33,819	33,819
Equilibrium cycle discharge (1/3 core), kg						
U^{234}	4.3	4.8	5.2	5.5	5.7	6.0
U^{235}	220.0	202.4	198.2	198.0	199.3	202.4
U^{236}	66.0	99.5	127.9	145.0	164.7	187.7
U^{238}	33,461	33,281	33,145	33,070	32,988	32,897
Fissile Pu	126.3	152.4	168.7	177.1	185.8	195.1

Table 11.4. Neutron Balances for Typical Cycles for PWR, Case 7

	Initial Balance, k = 1.002	Balance at End of Cycle					Equilibrium Cycle	
		1	2	3	4	5	Start	End
U^{234}	0.00159	0.00126	0.00136	0.00139	0.00134	0.00131	0.00147	0.00131
U^{235}	0.46932	0.23020	0.22711	0.22987	0.22962	0.22926	0.33251	0.22942
U^{236}	0.00026	0.00512	0.00643	0.00684	0.00627	0.00589	0.00342	0.00584
U^{238}	0.26781	0.27128	0.26289	0.26106	0.26487	0.26617	0.25945	0.26645
Np^{239}		0.00041	0.00038	0.00038	0.00039	0.00040		0.00040
Pu^{239}		0.22309	0.21261	0.20979	0.21335	0.21405	0.13045	0.21408
Pu^{240}		0.05072	0.05524	0.05429	0.05331	0.05313	0.02885	0.05302
Pu^{241}		0.02994	0.03923	0.03919	0.03676	0.03656	0.01489	0.03636
Pu^{242}		0.00069	0.00158	0.00191	0.00142	0.00139	0.00039	0.00137
Zr	0.01000	0.01029	0.00983	0.00974	0.00995	0.01003	0.00959	0.01004
H	0.06291	0.06574	0.06032	0.05930	0.06184	0.06272	0.05828	0.06289
O	0.00266	0.00269	0.00271	0.00271	0.00270	0.00270	0.00267	0.00270
B	0.15261						0.09044	
Xe^{135}		0.02096	0.02070	0.02064	0.02075	0.02079		0.02080
Sm^{149}		0.00698	0.00710	0.00708	0.00706	0.00701	0.00788	0.00701
Gross fission products		0.04735	0.05910	0.06238	0.05706	0.05530	0.02679	0.05500
Leakage	0.03285	0.03328	0.03341	0.03343	0.03321	0.03329	0.03293	0.03331
Total	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

Physics of the SSCR

The reference design for the spectral-shift-control reactor utilizes a batch-loaded fuel cycle with axial and radial enrichment zoning for power flattening. These enrichment zones were assumed to be homogenized to give an average core enrichment for the point model. Zero-dimensional depletion calculations were performed with 24 neutron energy groups for a nominal 30-year history with the assumption that the cycle time would be maintained constant throughout the history. This cycle time was treated as a parameter in optimizing fuel-cycle costs.

Spectral and cell calculations were performed to obtain effective 24-group microscopic cross sections for each of the nuclides present. Since the effective microscopic cross sections of the fertile material change appreciably (due to resonance self-shielding effects) as the moderator composition is changed, the spectral and cell calculations were repeated for end of life conditions (2% D_2O). The thorium broad-group absorption cross sections were then correlated as a function of the moderator composition.

The depletion calculations were performed by enriching the first clean batch cycle sufficiently to attain an initial excess reactivity of 0.5% at the assumed moderator composition. The system was depleted for some time increment sufficient to drop k_{eff} slightly below 1.0, and then the H_2O fraction was increased to give a reactivity of 1.005 at the

start of the next step. This procedure was continued until a 2% D_2O moderator composition (minimum concentration set by B & W) was reached. New multigroup microscopic thorium absorption cross sections were calculated at each moderator composition using the correlation mentioned above. The moderator composition during exposure is presented in Fig. 11.6 for the first cycle of a typical history.

The second and subsequent cycle calculations were performed in the same manner with the following assumptions:

1. The initial moderator composition was varied to keep the cycle time approximately the same for all cycles in the 30-year history;
2. Uranium was recycled from the previous cycle with 1% loss;
3. U^{233} from decay of Pa^{233} was recycled with 1% loss after a 180-day decay time;
4. U^{235} makeup was supplied as required to achieve the desired reactivity at the initial moderator composition.

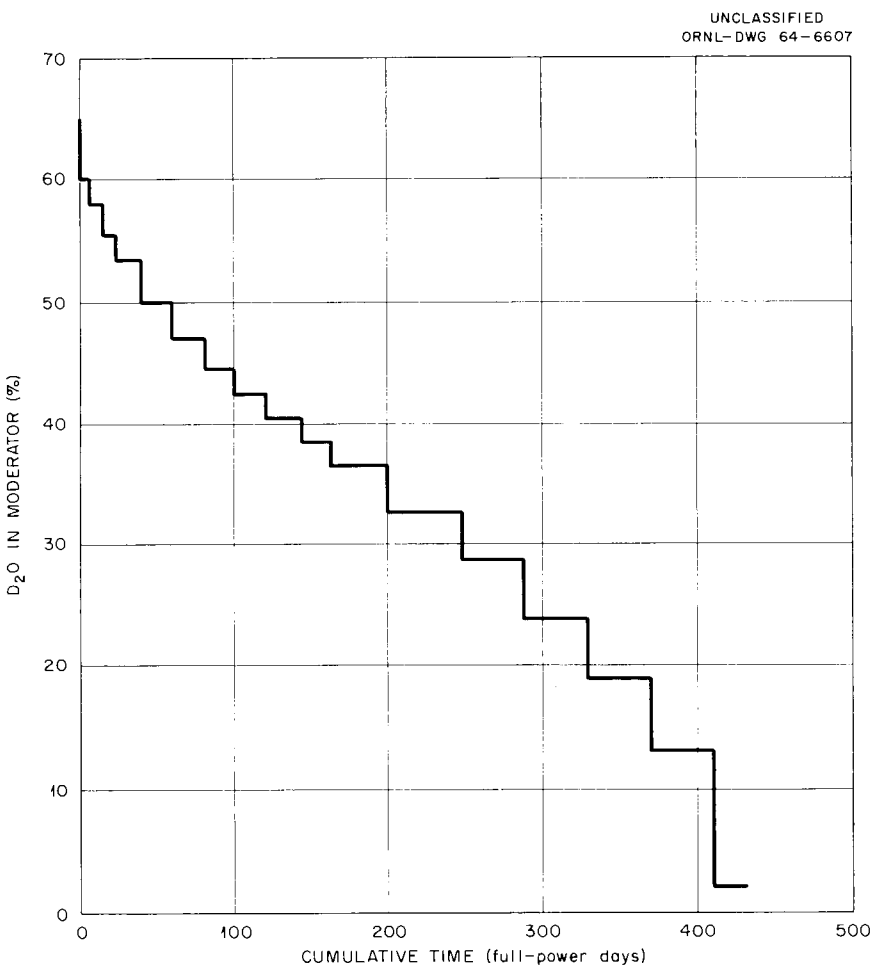

Fig. 11.6. D_2O Concentration During First Cycle of SSCR.

As in cycle one, makeup U^{235} requirements were based on an initial excess reactivity of 0.5%.

Depletion calculations were performed for 30-year histories (24 full-power years) with the cycle exposures varied from 22,000 to 42,000 Mwd/MT. Based on fuel-cycle cost calculations for these histories the optimum exposure was about 27,000 Mwd/MT of U + Th with the lower U^{233} eta value. Mass balances for representative cycles during the plant life are presented in Table 11.5. The conversion ratios were approximated by taking a linear average between the beginning and end of life values for each cycle.

The results given in Table 11.5 were obtained with the low eta (high alpha) cross-section set for U^{233}. Results based on the high eta (low alpha) set are shown in Table 11.6. The B & W results, shown for comparison in Table 11.7, are based on cross sections similar to the high eta set. Since these results were obtained, conversations between ORNL and B & W have led to changes in the Doppler-broadened thorium cross sections used by both teams so that reasonable agreement now seems to exist as to the initial D_2O concentration for a given fuel enrichment.

Comparison of the first four cycles of Table 11.6 with Table 11.7 shows that the average conversion ratios calculated by ORNL are only a few points lower than those of B & W. Whether this discrepancy has been eliminated by the changes made in Doppler-broadened thorium cross sections has not yet been determined.

Table 11.5. SSCR Mass Balances for Case C-3

Minimum fuel-cycle cost case, lower eta for U^{233}

Cycle number	1	2	3	4	10
Conversion ratio[a]	0.64	0.68	0.69	0.69	0.67
Cycle time, full-power days	643	618	607	644	616
Exposure, Mwd/MT of U + Th	31,100	29,700	29,000	30,600	28,800
Initial D_2O concentration, %	71	73	72	71	67
Initial loading, kg					
Thorium	63,833	63,833	63,833	63,833	63,833
U^{233}	0	987	1292	1414	1558
U^{234}	36	175	324	476	660
U^{235}	3389	2194	1983	2001	2139
U^{236}	8.9	466	664	796	1263
U^{238}	181	195	212	229	279
Fissile material, wt %	5.02	4.69	4.79	4.97	5.30
Discharged fuel, kg					
Thorium	62,042	62,139	62,159	62,162	62,320
U^{233}	997	1305	1428	1490	1577
U^{234}	166	316	429	509	665
U^{235}	1229	944	927	979	1179
U^{236}	468	668	801	912	1316
U^{238}	143	157	172	186	232
Pu^{239}	9.4	11	12	13	16
Pu^{241}	3.6	3.8	4.1	4.4	4.7
Fissile U, wt %	3.42	3.43	3.57	3.73	4.10

[a]Linear average of conversion ratios at beginning and end of cycle.

Table 11.6. SSCR Mass Balances for Case B-3

Minimum fuel-cycle cost case,
higher eta for U^{233}

Cycle number	1	2	3	4	10
Conversion ratio[a]	0.64	0.72	0.73	0.74	0.745
Cycle time, full-power days	693	546	528	527	472
Exposure, Mwd/MT of U + Th	33,600	26,400	25,500	25,400	22,600
Initial D_2O concentration, %	71	73	72	71	67
Initial loading, kg					
Thorium	63,833	63,833	63,833	63,833	63,833
U^{233}	0	1004	1239	1322	1394
U^{234}	36	177	300	389	561
U^{235}	3338	1767	1476	1428	1374
U^{236}	8.9	480	615	691	920
U^{238}	181	174	179	185	199
Fissile material, wt %	4.95	4.11	4.01	4.05	4.05
Discharged fuel, kg					
Thorium	61,892	62,262	62,334	62,356	62,543
U^{233}	1015	1252	1336	1372	1406
U^{234}	172	296	386	448	566
U^{235}	1143	797	732	743	808
U^{236}	483	620	696	755	946
U^{238}	141	143	149	155	171
Pu^{239}	9.3	9.0	9.3	9.7	10
Pu^{241}	3.7	3.1	3.0	3.0	2.7
Fissile U, wt %	3.33	3.13	3.15	3.21	3.33

[a]Linear average of conversion ratios at beginning and end of cycle.

Table 11.7. B & W SSCR Mass Balances

Higher eta for U^{233}

Cycle number	1	2	3	4
Conversion ratio[a]	0.67	0.74	0.76	0.76
Cycle time, full-power days	675	600	650	675
Exposure, Mwd/MT of U + Th	33,200	29,500	32,000	33,200
Initial D_2O concentration, %	56	61	63	61
Initial loading, kg				
Thorium	62,954	62,898	62,641	62,395
U^{233}		1023	1305	1426
U^{234}		111	223	315
U^{235}	3230	1782	1520	1474
U^{236}		393	531	624
U^{238}	243	225	235	244
Fissile U, wt %	4.86	4.22	4.25	4.36
Discharge fuel, kg				
Thorium	60,778	61,175	60,796	60,514
U^{233}	1043	1332	1354	1507
U^{234}	114	228	322	386
U^{235}	1247	845	838	743
U^{236}	401	542	637	713
U^{238}	186	186	192	199
Pu^{239}	15	15	16	17

[a]Integrated average.

Physics of the HWR

Uranium-Fueled HWR (HWR-U)

The uranium-fueled HWR was designed for shutdown refueling with one-third of the core being replaced each time. The fuel management program specified an initial loading having two radial enrichment zones. The outer zone, comprising one-third the core volume, would be loaded with slightly enriched uranium; the inner zone (two-thirds the core volume) would be fueled with natural uranium. At equilibrium the fuel management would be a three-batch scatter-reloading program, with slightly enriched uranium being fed. This was simulated in the point depletion calculation by treating three batches of fuel exposed to the same flux. At startup, two of these batches were natural uranium and one was slightly enriched.

Point multigroup depletion calculations were made over a range of variables for heavy-water reactors. The calculated dependence of exposure on feed enrichment in a uranium-loaded HWR for both the first cycle and equilibrium is shown in Fig. 11.7. In a three-batch, partially reloaded core without recycle, equilibrium is approached after about four cycles. One-dimensional depletion calculations were also made, for the

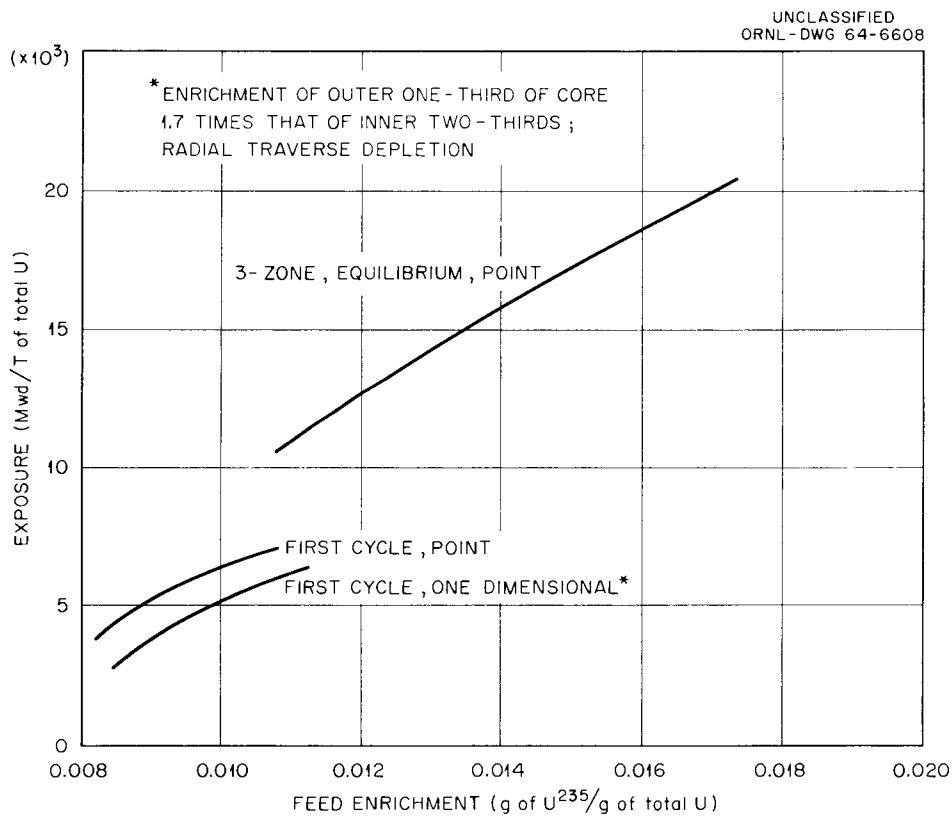

Fig. 11.7. Exposure Calculated for the HWR-U.

first cycle, to determine the behavior of a core with higher enriched fuel in an outer zone than toward the center. This calculation used the same set of microscopic cross sections as that employed in the point depletion calculations (20 fast, 4 thermal groups) for recycle. The effect of low reactivity of fuel in the central part of the core in reducing the achievable exposure is indicated by the lowest curve in Fig. 11.7. However, effects of early cycles on 30-year economics are slight when cycle times are short, as for these cores.

Special attention was given to satisfying experimental determinations of reactivity. Allowance was made for higher fast fission in U^{238} than calculated for a homogenized cell, relatively high mean chord lengths were used in resonance calculations to allow for shielding of the source in inner rings of concentric tube elements, and relatively low core bucklings were used to compensate for overestimates of loss to structural materials obtained with THERMOS cell calculations.

For a point calculation of depletion of a uranium-loaded HWR, initial zone and feed enrichments were specified and control poison was adjusted to effect a critical situation on the average through each cycle. A summary of results is presented in Table 11.8. Selected neutron balances are given in Tables 11.9 and 11.10. Mass balances are listed in Table 11.11. Results obtained from Du Pont indicated that their calculations would predict an equilibrium cycle exposure of 15,400 Mwd/MT with 1.2% feed enrichment compared with 12,500 Mwd/MT from ORNL results. The cause of this difference in results has not been resolved at this time. The information provided by Du Pont did not include sufficiently detailed information on the fuel cycles (such as neutron and mass balances) to permit a complete comparison of results to be made.

The fuel element for this reactor consists of three concentric fuel tubes. Shifts may be expected to occur in the power distribution among these three tubes as a result both of nonuniform buildup of plutonium and of nonuniform burnout of U^{235}, both of which occur preferentially in the outer tube. Preliminary calculations, based on a slab model of the element, indicate the strong possibility of a net shift in power toward the outer tube. Calculations made by Wade at Savannah River indicate that the power may shift toward either the inner or outer tube, depending on the design, but that the effect is small. It would be very desirable to investigate this question further by more elaborate calculations, such as Monte Carlo calculations of the U^{238} capture distribution, coupled with depletion calculations.

Thorium-Loaded HWR (HWR-Th)

The thorium-loaded HWR was designed for on-power continuous refueling in order to avoid reactivity transients associated with decay of Pa^{233} to U^{233} during shutdowns. On-power refueling of an entire fuel channel was approximated by performing two sets of calculations. In each case it was assumed that there were six batches of fuel exposed to the same flux, and full recycle of uranium (after losses) was assumed.

The reference design provided by Du Pont (described in Chap. 3) included 88 lattice positions in the radial reflector in which blanket elements of ThO_2 would be located. This blanket could not be treated in the

Table 11.8. Summary of Physics Calculations for HWR-U

Case	E-1	E-2	E-3	E-4	E-5
First-cycle data					
Initial k (no rod)	1.124	1.138	1.165	1.895	1.2122
Cycle time, full-power days	78.4	85.6	99.1	114.0	124.5
Discharge exposure, Mwd/MT	4307	4703	5445	6263	6840
Initial enrichment, wt % U^{235}	0.843	0.872	0.929	0.987	1.045
Equilibrium cycle					
Initial k (no rod)	1.1142	1.1256	1.1458	1.1633	1.1770
Cycle time, full-power days	67.6	76.1	91.9	106.6	121.8
Discharge exposure, Mwd/MT	11,142	12,540	15,150	17,570	20,080
Feed enrichment, wt % U^{235}	1.11	1.19	1.36	1.53	1.71
Discharge enrichment, wt % U^{235}	0.246	0.235	0.220	0.199	0.194
Discharge Pu, g/kg of U feed	4.367	4.553	4.824	5.024	5.197
Pu composition, wt %					
Pu^{239}	62.93	61.24	59.00	57.46	56.19
Pu^{240}	22.14	22.23	22.04	21.67	21.25
Pu^{241}	11.23	12.08	13.18	13.92	14.46
Pu^{242}	3.70	4.45	5.78	6.95	8.10

Table 11.9. Neutron Balances for Uranium Loaded HWR-U at End of Equilibrium Cycle

Case	E-1	E-2	E-3	E-4	E-5
H	0.006401	0.006202	0.005879	0.005598	0.005367
D	0.007565	0.007457	0.007275	0.007154	0.007057
U^{235}	0.213863	0.209166	0.203634	0.200491	0.196623
U^{236}	0.002288	0.002554	0.003065	0.003581	0.004107
U^{238}	0.294067	0.287899	0.277742	0.269836	0.263309
Pu^{239}	0.242335	0.243205	0.242913	0.240995	0.239102
Pu^{240}	0.049911	0.052764	0.056650	0.059408	0.061665
Pu^{241}	0.032565	0.036063	0.041459	0.045228	0.048481
Pu^{242}	0.000597	0.000745	0.001032	0.001308	0.001601
Oxygen	0.003142	0.003144	0.003144	0.003173	0.003198
Zr + Nb	0.057192	0.055765	0.053439	0.051519	0.049941
Xe^{135}	0.023705	0.023665	0.023567	0.023715	0.023836
Sm^{149}	0.006971	0.007176	0.007494	0.007803	0.008056
Gross fission products	0.048460	0.053261	0.061782	0.069277	0.076752
Leakage	0.010938	0.010934	0.010925	0.010914	0.010905
Total	1.0	1.0	1.0	1.0	1.0

Table 11.10. Neutron Balance for HWR-U Case E-3

Equilibrium feed enrichment = 1.36%

	Start Cycle 1	End Cycle 1	End Cycle 2	End Cycle 3	End Cycle 4	End Cycle 5	End Cycle 6	End Equilibrium Cycle
H	0.007742	0.006887	0.006211	0.005889	0.005876	0.005859	0.005879	0.005879
D	0.008875	0.007848	0.007479	0.007305	0.007286	0.007299	0.007275	0.007275
U^{235}	0.549426	0.221130	0.205625	0.205548	0.202955	0.203246	0.203714	0.203634
U^{236}		0.001780	0.002433	0.003051	0.003075	0.003082	0.003064	0.003065
U^{238}	0.351025	0.309612	0.288636	0.278692	0.278005	0.278027	0.277730	0.277742
Pu^{239}		0.248277	0.246240	0.241160	0.243641	0.242448	0.242839	0.242913
Pu^{240}		0.041676	0.053961	0.056035	0.056817	0.056952	0.056616	0.056650
Pu^{241}		0.020737	0.036267	0.040650	0.041139	0.041397	0.041457	0.041459
Pu^{242}		0.000194	0.000674	0.001045	0.000992	0.001037	0.001038	0.001032
Oxygen	0.003583	0.003151	0.003158	0.003164	0.003154	0.003172	0.003144	0.003144
Zr + Nb	0.068423	0.060719	0.055875	0.053582	0.053454	0.053393	0.053438	0.053439
Xe^{135}		0.023947	0.023774	0.023735	0.023650	0.023791	0.023566	0.023567
Sm^{149}		0.006541	0.007168	0.007525	0.007533	0.007550	0.007493	0.007494
Gross fission products		0.036576	0.051574	0.061694	0.061498	0.061822	0.061822	0.061782
Leakage	0.010925	0.010925	0.010925	0.010925	0.010925	0.010925	0.010925	0.010925
Total	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

Table 11.11. Mass Balance Over Life of HWR-U

Case	E-1	E-2	E-3	E-4	E-5
History time, years	18.5	20.0	20.1	20.1	20.4
Initial loading, kg					
U^{235}	575.5	595.4	635.3	675.1	714.9
U^{238}	67,714	67,714	67,714	67,714	67,714
Total feed material, kg					
U^{235}	20,506	21,282	20,293	19,674	19,515
U^{238}	1,850,848	1,783,134	1,489,707	1,286,564	1,151,136
Total material removal, kg					
U^{235}	4674	4319	3413	2848	2447
U^{236}	2484	2655	2631	2612	2637
U^{238}	1,833,854	1,765,238	1,472,578	1,270,159	1,135,041
Fissile Pu	6000	5956	5193	4623	4238
Equilibrium cycle feed (1/3 core), kg					
U^{235}	252.3	272.2	312.0	351.8	391.7
U^{238}	22,571	22,571	22,571	22,571	22,571
Equilibrium cycle discharge (1/3 core), kg					
U^{235}	55.1	52.6	49.1	46.8	44.1
U^{236}	30.9	34.4	41.0	47.3	53.6
U^{238}	22,359	22,339	22,304	22,274	22,243
Fissile Pu	74.0	76.3	79.7	82.2	84.3

point depletion model, so all results reported here apply to the reference reactor without the blanket. The blanket production of U^{233} could be estimated from the core leakage fraction.

Previous comparisons of THERMOS results with experimental data indicated that THERMOS overestimates the flux depression in the fuel of these large multiring elements. This causes the relative neutron absorptions in structure (pressure tubes) to be overestimated. To compensate for this recognized neutron loss, the leakage in the HWR calculations was made low. The overall reactivity balance should still be valid, since the sum of leakage and parasitic captures has been maintained at the value thought to be correct. Hence, when the neutron leakage is used to estimate blanket production of U^{233}, the value should be about 2% rather than the 1.1% given in Table 11.12.

The first treatment assumed that when a batch of fuel was discharged and replaced with recycle fuel, only enough makeup U^{235} to make the system critical would be supplied. During depletion, more U^{235} would be added as needed to maintain criticality. This treatment should be a reasonable approximation to the on-stream refueling situation. For purposes of the economics, it was assumed that all the makeup U^{235} needed during a cycle was bought at the start of the cycle. The cycle time was treated as a parameter in fuel cost optimization. Results from this treatment are presented in Table 11.13 as cases A-3, A-4, and A-5.

In order to place a lower limit on exposure at a given feed enrichment, the thorium-loaded HWR was also calculated as a six-batch scatter-reloaded core with poison control. These results are presented in Table 11.13 as cases A-1 and A-2.

Table 11.12. Neutron Balance for HWR-Th Case A-3

Nuclide	Initial	End of Cycle						Final
		1	2	3	4	5	10	
Hydrogen	0.00683	0.00609	0.00598	0.00591	0.00586	0.00583	0.00587	0.00535
D	0.00788	0.00749	0.00742	0.00738	0.00736	0.00733	0.00735	0.00706
Th^{232}	0.43346	0.39108	0.38332	0.37893	0.37550	0.37279	0.37482	0.34638
Pa^{233}		0.01133	0.01227	0.01223	0.01211	0.01201	0.01215	0.01083
U^{233}		0.05592	0.11589	0.16617	0.20572	0.23660	0.32328	0.33143
U^{234}	0.00108	0.00178	0.00282	0.00404	0.00540	0.00684	0.01513	0.04207
U^{235}	0.49071	0.42419	0.35855	0.30249	0.25834	0.22390	0.12763	0.11819
U^{236}	0.00015	0.00127	0.00205	0.00270	0.00325	0.00372	0.00506	0.01039
U^{238}	0.00167	0.00207	0.00218	0.00226	0.00234	0.00241	0.00256	0.00374
Pu^{239}		0.00098	0.00148	0.00174	0.00188	0.00198	0.00207	0.00300
Pu^{240}		0.00014	0.00033	0.00045	0.00053	0.00058	0.00059	0.00086
Pu^{241}		0.00003	0.00014	0.00025	0.00034	0.00041	0.00040	0.00058
Pu^{242}						0.00001	0.00001	0.00001
Oxygen	0.00315	0.00317	0.00317	0.00316	0.00316	0.00316	0.00316	0.00316
Zr	0.04379	0.03557	0.03501	0.03471	0.03446	0.03427	0.03446	0.03197
Xe^{135}		0.02343	0.02305	0.02275	0.02251	0.02233	0.02185	0.02160
Sm^{149}		0.00522	0.00569	0.00593	0.00604	0.00606	0.00554	0.00536
Gross fission products		0.01896	0.02937	0.03762	0.04392	0.04849	0.04679	0.04674
Leakage	0.01128	0.01128	0.01128	0.01128	0.01128	0.01128	0.01128	0.01128
Total	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

Table 11.13. Summary of Results for HWR-Th

Case	A-1	A-2	A-3	A-4	A-5	G-4
First cycle						
Initial k (no rod)	1.045	1.065	1.005	1.005	1.005	1.005
Cycle time, full-power days	12.5	25	75	100	138	100
Initial enrichment, wt %	1.51	1.58	1.76	1.81	1.87	1.81
Final cycle						
Initial k (no rod)	1.045	1.065	1.005	1.005	1.005	1.005
Cycle time, full-power days	58	83	75	100	138	100
Number of cycles	60	60	60	60	60	60
Exposure, Mwd/MT (U + Th)	16,000	23,000	21,000	27,800	38,200	28,760
U^{235} makeup enrichment, wt %	0.429	0.654	0.407	0.652	1.111	0.623
U^{235} recycle enrichment, wt %	0.190	0.258	0.165	0.175	0.179	0.164
U^{233} recycle enrichment, wt %	1.333	1.306	1.335	1.319	1.293	1.309
Total fissile feed enrichment, wt %	1.952	2.218	1.907	2.146	2.583	2.096

Selected neutron balances are given in Table 11.12 and mass balances in Table 11.14. Case G-4 is the same as A-4 except that the higher eta U^{233} cross sections were used.

For the equilibrium recycle case, Du Pont results indicated that a total feed fuel enrichment about 2.0 to 2.1% fissile would produce about 23,000 Mwd/MT. ORNL results for the continuous fuel feed model predict that an enrichment about 2.0% fissile would be required to achieve this exposure. Results based on poison control predict an enrichment of 2.2%. This comparison indicates that ORNL results for both calculational models are in good agreement with Du Pont results. The fuel feed model should be the better approximation to continuous on-power refueling, and those results should be taken more seriously. The poison control results should place a conservative lower limit on the exposures to be obtained.

Physics of the HTGR

There are a number of fuel management schemes that might be considered for the high-temperature gas-cooled reactor. With its coated particle fuel, the feed, recycle, and bred fuel can be kept separated by using particles of different sizes. Thus long-burnup uranium may be discarded if economically desirable. If U^{233} could be sold at a premium price to another reactor, then bred fuel might be sold after processing. If U^{233} were worth as much or more in the HTGR as it could be sold for, then continued recycle of bred fuel might be desirable. In this study emphasis was placed on full recycle, but a number of partial recycle modes were also considered. The choice of recycle mode would have to be based on economic considerations, as discussed in Chapter 12.

The initial information provided for the HTGR reference design specified a six-batch scatter-reloading fuel managment program. Subsequently, this was revised to a 12-batch scatter-reloading program. The calculations reported here are for the 12-batch core, with one-twelfth of the core being removed at each shutdown. This was treated in the TONG program by following 12 sets of nuclides exposed to a 15-group flux. The calculated control poison concentration through the first cycle of two case histories is shown in Fig. 11.8. The specified initial reactivity (with control poison removed) was achieved by adjustment of the feed uranium concentrations; required core loadings of U^{235} are shown. A one-dimensional radial traverse depletion calculation was also made with the core zoned as recommended by General Atomic and with an average U^{235} concentration estimated to yield a 2-year core life. The poison concentration calculated through the life is plotted in Fig. 11.9. Comparison of the results shows that the point calculation required a loading of 3700 kg U^{235} to achieve 19,500 EFPH, while the one-dimensional calculation required somewhat less fuel, 3500 kg U^{235} for 19,000 EFPH.

In order to estimate the effect of uncertainties in the U^{233} epithermal cross sections, selected cases were rerun using a higher eta for U^{233} (similar to the GA "modified" set). Calculations were also made based on the BeO spine replaced with graphite to eliminate the high-cost beryllium from the moderator.

Table 11.14. Mass Balance Over Life of HWR-Th

Case	A-1	A-2	A-3	A-4	A-5	G-4
History time, years	11.2	19.6	15.4	20.5	28.2	20.5
Last cycle time, years	0.159	0.229	0.205	0.274	0.378	0.274
Initial loading, kg						
U^{234}	12.97	13.54	15.13	15.59	16.12	15.54
U^{235}	1,202.6	1,255.7	1,402.7	1,445.2	1,494.9	1441.0
U^{236}	8.38	8.75	9.78	10.07	10.42	10.05
U^{238}	97.7	102.0	114.0	117.4	121.5	117.1
Th^{232}	78,136	78,136	78,136	78,136	78,136	78,136
Total feed material (excludes that recycled), kg						
U^{234}	46.3	85.17	52.9	72.9	110.5	70.2
U^{235}	4,290	7,896	4,905	6,763	10,245	6,513
U^{236}	29.9	55.04	34.2	47.1	71.4	45.4
U^{238}	348.6	641.5	398.4	549.4	832.3	529.2
Th^{232}	109,582	115,997	113,756	117,954	123,657	118,095
Total material removed, kg						
U^{233}	10,548	11,021	11,260	11,370	11,395	11,172
U^{234}	3,554	4,596	4,424	4,900	5,308	4,772
U^{235}	3,184	2,899	1,890	1,848	1,880	1,765
U^{236}	3,748	5,979	4,348	5,450	7,197	5,280
U^{238}	1,692	2,399	1,779	2,119	2,698	2,051
Th^{232}	834,599	827,667	830,084	825,519	819,326	825,362
Fissile Pu	39.53	68.11	45.45	58.40	80.00	56.10
Total material recycled, kg						
U^{233}	9,191	9,669	9,892	10,004	10,039	9,925
U^{234}	2,928	3,889	3,734	4,180	4,571	4,073
U^{235}	2,824	2,419	1,517	1,410	1,345	1,348
U^{236}	3,124	4,926	3,642	4,532	5,952	4,396
U^{238}	1,443	2,003	1,520	1,790	2,252	1,735
Th^{232}	736,888	730,473	732,713	728,516	722,813	728,374
Last cycle feed (excluding recycle), kg						
U^{234}	0.62	0.96	0.59	0.95	1.64	0.86
U^{235}	57.9	89.0	55.0	88.5	152.2	84.4
U^{236}	0.41	0.62	0.38	0.61	1.06	0.64
U^{238}	4.70	7.22	4.46	7.19	12.37	6.90
Th^{232}	341.9	439.0	391.2	462.5	558.0	465.6
Last cycle recycle feed, kg						
U^{233}	179.8	177.8	180.1	179.0	1772.2	177.4
U^{234}	84.0	94.1	92.0	95.8	97.5	93.1
U^{235}	25.6	35.2	22.3	23.7	24.5	22.3
U^{236}	79.7	140.3	89.5	116.5	158.3	112.1
U^{238}	29.9	50.5	31.8	40.6	54.7	38.9
Th^{232}	12,681	12,584	12,630	12,560	12,465	12,557
Last cycle one zone discharge, kg						
U^{233}	181.6	179.7	181.9	180.8	179.0	179.2
U^{234}	86.3	95.8	93.6	97.1	98.6	94.3
U^{235}	59.6	49.7	22.9	24.1	24.9	22.6
U^{236}	91.8	145.7	89.3	119.2	162.2	114.9
U^{238}	37.5	53.4	31.9	40.8	55.6	39.1
Th^{232}	12,810	12,711	12,759	12,687	12,591	12,684
Fissile Pu	0.97	1.63	0.86	1.18	1.30	1.23

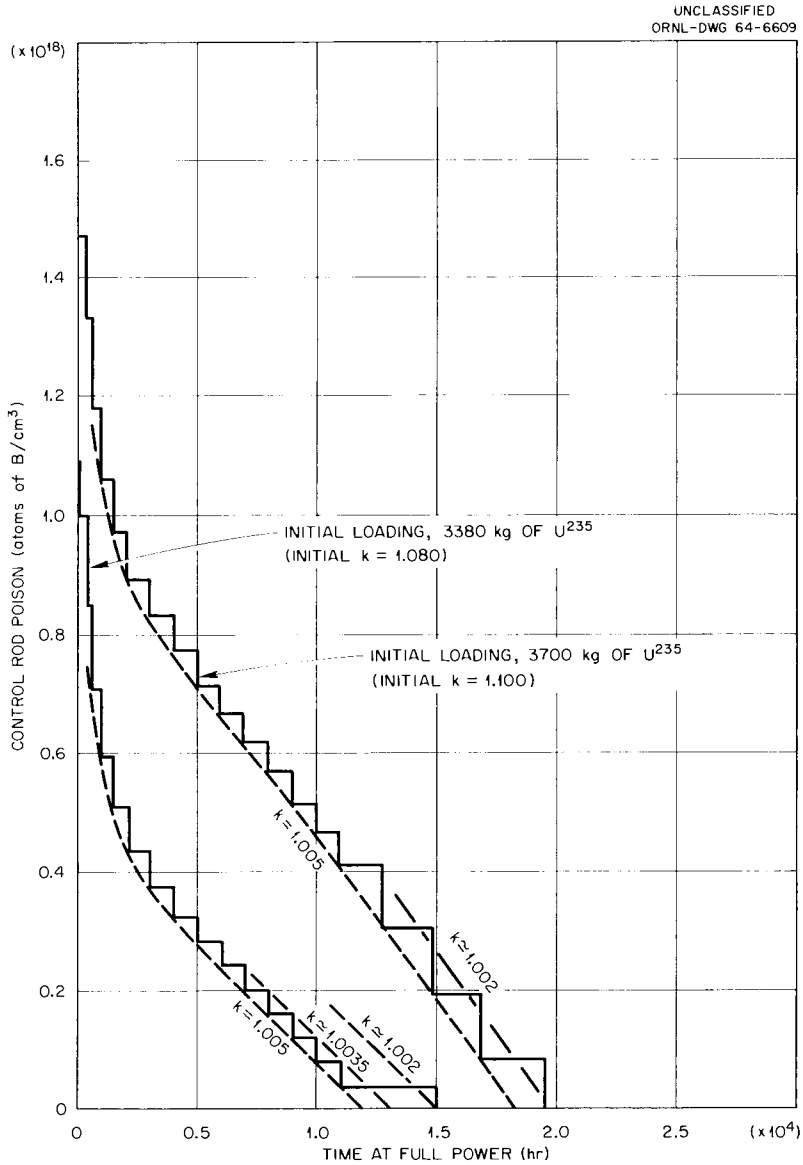

Fig. 11.8. Control Poison Concentration Through First Cycle of Two Case Histories for HTGR.

Results from all the calculations are summarized in Tables 11.15 and 11.16. The case identifications refer to specific design and fuel-management conditions, as explained below.

For cases 1, 8, and 9, it was considered that all the uranium available for recycle was returned to the reactor. None of the uranium was discarded or sold. Case 2 is the same as the GA "type I recycle." In this case, none of the feed U^{235} (initial loading and makeup) is recycled; it is assumed to be sold or discarded after only one pass through the reactor. The uranium produced by thorium captures is assumed to be recycled only once and then sold. Case 3 is the same as the GA "type II

Table 11.15. Summary of Cycle Performance for HTGR Based on Lower U^{233} Eta

	Full recycle of all uranium including feed U^{235}			GA Type I, Case 2	GA Type II, Case 3	Once-through recycle of bred fuel and then discharge; initial core loading recycled once; makeup U^{235} not recycled; same as Type I in equilibrium					
	Case 1	Case 8	Case 9			4	10	5	B-10	B-5	B-55
First cycle											
Initial k	1.0925	1.0925	1.0925	1.0925	1.0925	1.08	1.086	1.0925	1.086	1.0925	1.10
Cycle time, full-power days	730	730	730	725	725	630	639	725	505	640	659
Average exposure, Mwd/MT of U + Th	16,650	16,650	16,650	16,540	16,540	14,400	14,580	16,540	13,296	16,876	17,381
Initial fissile enrichment, wt % U^{235}	3.59	3.59	3.59	3.58	3.58	3.40	3.49	3.58	3.24	3.32	3.42
Initial Be-to-Th ratio	26.14	26.14	26.14	26.14	26.14	26.14	26.14	26.14	29.9	29.9	29.9
Equilibrium or final cycle											
Initial k	1.0467	1.040	1.04335	1.0467	1.0467	1.040	1.0433	1.0467	1.0433	1.0467	1.05
Cycle time, full-power days	269	155	198	198	236	101	155	198	104	151	192
Discharge exposure, Mwd/MT of U + Th	75,600	45,000	57,105	58,480	68,630	30,140	48,500	58,500	30,145	51,174	64,540
Initial enrichment, wt %											
Feed U^{235}	3.53	1.11	1.89	1.75	2.14	0.96	1.21	1.69	0.95	1.30	1.09
Recycle U^{233}	2.37	2.23	2.26	2.01	2.24	1.67	1.91	2.02	1.74	1.91	2.03
Recycle U^{235}	0.66	0.58	0.54	0.10	0.32	0.04	0.07	0.09	0.04	0.08	0.11
Discharge enrichment, wt %											
For recycle	3.45	3.07	3.10	2.27	2.85	1.78	2.11	2.27	1.88	2.14	2.33
For sale or discard				0.50	0.19	0.66	0.52	0.52	0.54	0.39	0.41
Ratio of final Th + U to initial Th + U	0.902	0.947	0.928	0.937	0.921	0.967	0.950	0.938	0.997	0.943	9.28
Conversion ratios											
Average gross, before losses	0.741	0.840	0.805	0.852	0.809	0.922	0.883	0.845	0.909	0.870	0.830
Average net (less processing losses)	0.739	0.836	0.802	0.849	0.807	0.913	0.879	0.843	0.906	0.868	0.828
Average net (less processing losses and fuel discard)					0.791	0.782	0.800	0.797	0.811	0.837	0.792

Table 11.16. Summary of Cycle Performance for HTGR Based on Higher U^{233} Eta

	Full Recycle, A-8	GA Type I, A-2	GA Type II, A-3	Once-through recycle of bred fuel and then discharge; initial core loading recycled once; makeup U^{235} not recycled; same as type I in equilibrium				
				A-4	A-10	A-5	AB-5	AC-10
First cycle								
Initial k	1.0925	1.0925	1.0925	1.08	1.086	1.0925	1.0925	1.10
Cycle time, full-power days	886.4	880.3	880.3	715	800	886	702	378
Average exposure, Mwd/MT of U + Th	20.213	20.075	20.075	16,326	18,258	20,210	18,349	15,007
Initial fissile enrichment, wt % U^{235}	3.59	3.58	3.58	3.40	3.49	3.59	3.32	4.54
Initial Be-to-Th ratio	26.14	26.14	26.14	26.14	26.14	26.14	29.87	0
Equilibrium cycle								
Initial k	1.040	1.0407	1.0467	1.04	1.0433	1.0467	1.0467	1.05
Cycle time, full-power days	148.5	201.2	232	103	153	199	151	72.0
Discharge exposure, Mwd/MT	43,706	60,039	67,794	30,890	45,945	58,863	51,464	37,357
Initial enrichment, wt %								
Feed U^{235}	0.89	1.52	1.75	0.83	0.64	0.15	1.20	1.71
Recycle U^{233}	2.17	1.98	2.15	1.67	1.87	1.98	1.87	1.91
Recycle U^{235}	0.49	0.15	0.30	0.04	0.07	0.09	0.07	0.04
Discharge enrichment, wt %								
For recycle	2.79	2.24	2.70	1.78	2.05	2.23	2.07	2.05
For sale or discard		0.48	0.146	0.30	0.46	0.37	0.41	0.93
Ratio of final Th + U to initial Th + U	0.954	0.940	0.923	0.966	0.956	0.936	0.946	0.962
Conversion ratios								
Average gross, before losses	0.876	0.866	0.838	0.942	0.910	0.870	0.903	0.831
Average net (less processing losses)	0.871	0.864	0.835	0.938	0.907	0.868	0.892	0.828
Average net (less processing losses and fuel discard)		0.822	0.822	0.841	0.843	0.829	0.841	0.696

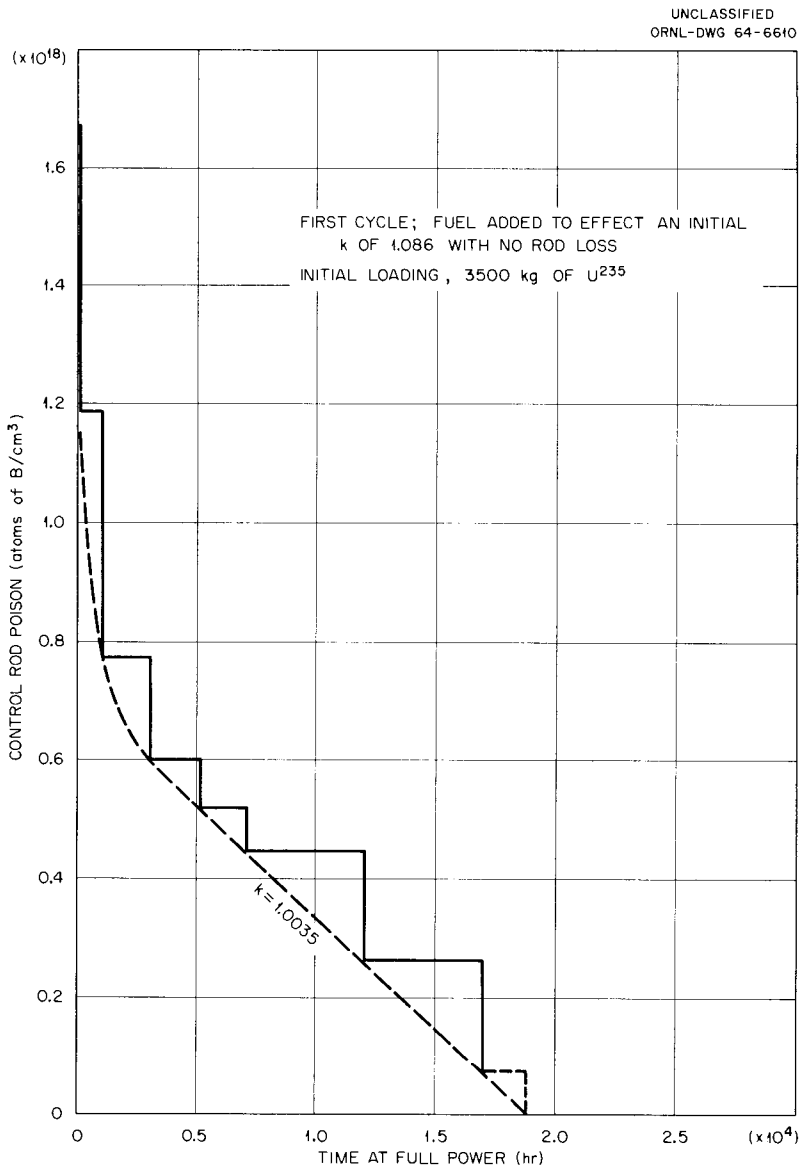

Fig. 11.9. Depletion of HTGR in a Radial Traverse.

recycle," in which none of the bred uranium is sold and none of the feed U^{235} is recycled.

For cases 4, 5, 10, and 55, it was considered that (1) the initial loading of U^{235} would be recycled once and then sold, but U^{235} makeup to later cycles would not be recycled, and (2) bred uranium would be recycled only once and then sold. This is the same as type I recycle, except that the initial loading is recycled once before being sold.

The prefixes on the case numbers identify that the following fuel designs or nuclear data were used:

1. No prefix — reference design as submitted by GA; Be-to-Th ratio of 27.8; lower eta for U^{233};
2. B cases — Be-to-Th ratio changed to 32;
3. A cases — changed to higher eta for U^{233};
4. C cases — fuel element changed to all-carbon moderator.

Combinations of the letter prefixes indicate that more than one variable was changed at the same time.

Conversion ratios calculated by ORNL have been compared with values reported by GA. ORNL values are 0.02 to 0.05 lower than those of GA. The source of this difference has not been thoroughly investigated, but it is probably associated with small differences in cross sections, fission-product treatments, and calculational models.

Neutron balances for case 5 are given in Table 11.17 and for case 10 in Table 11.18. Mass balances are given in Table 11.19.

Physics of the SGR

The reference design for the sodium-graphite reactor calls for a six-batch fuel-management program in which one-sixth of the fuel elements would be discharged at each shutdown. In addition, about 50 fuel elements would be moved from positions near the periphery to positions near the center of the core. In the point-depletion calculations performed in this study the six batches of fuel were exposed uniformly to the same flux. The neutron energy spectrum was described by a 26-group treatment with 20 groups above 0.876 ev and six groups in the thermal range. At the end of each cycle the appropriate batch of fuel was discharged and reloaded with fresh fuel to achieve a specified reactivity for the start of the next cycle.

The cycle time was treated as a parameter by varying the value specified for the initial reactivity at the start of a cycle. Feed enrichment to achieve this reactivity was determined by a direct eigenvalue calculation. During depletion the system was kept critical by introduction of a uniform poison, with the concentration being calculated as a function of time.

In comparing ORNL and AI physics results, it was discovered that the AI physics calculations were not based on the same reactor as the reference design presented to ORNL. The principal differences are listed in Table 11.20.

All the physical differences in reactor description would lead to higher enrichments (for a given exposure) being required for the reference design on which the ORNL calculations were based than for the reactor analyzed by AI. The larger number of control rod positions (with stainless steel process tubes) and the stainless steel support tube both have the effect of putting more stainless steel into the system and thus increasing parasitic losses to structural materials. The most significant difference in system designs is the use of sodium-cooled control rods rather than dry rods, which increases the enrichment required for a given exposure by about 0.22%.

The equilibrium cycle exposures are shown as a function of feed enrichment in Fig. 11.10 for both wet and dry control rods. Since earlier

Table 11.17. Neutron Balances for Typical Cycles for HTGR Case 5

	Initial Balance, k = 1.005	Balance at End of Cycle						
		1	2	4	6	10	20	44
Th^{232}	0.36177	0.36382	0.35818	0.34779	0.34080	0.33244	0.33285	0.34312
Pa^{233}		0.00846	0.00832	0.00815	0.00795	0.00755	0.00751	0.00800
U^{233}		0.18620	0.20078	0.23308	0.26709	0.29104	0.31477	0.32633
U^{234}	0.00240	0.00733	0.00866	0.01160	0.01538	0.02177	0.03129	0.03326
U^{235}	0.47706	0.25720	0.24034	0.20364	0.16486	0.14012	0.11468	0.10068
U^{236}	0.00019	0.00823	0.00888	0.01038	0.01187	0.01384	0.01157	0.00506
U^{238}	0.00421	0.00386	0.00384	0.00381	0.00379	0.00400	0.00310	0.00148
Pu^{238}		0.00024	0.00039	0.00077	0.00121	0.00175	0.00222	0.00065
Pu^{239}		0.00383	0.00377	0.00321	0.00245	0.00113		
Pu^{240}		0.00146	0.00142	0.00126	0.00098	0.00045		
Pu^{241}		0.00108	0.00118	0.00119	0.00098	0.00046		
Pu^{242}		0.00004	0.00001	0.00001	0.00009	0.00007		
Be	0.06706	0.06739	0.06753	0.06791	0.06817	0.06785	0.06781	0.06825
B	0.06003							
C	0.00429	0.00450	0.00456	0.00469	0.00470	0.00444	0.00440	0.00477
O	0.00272	0.00272	0.00273	0.00274	0.00275	0.00275	0.00275	0.00275
Xe^{135}		0.01343	0.01343	0.01352	0.01348	0.01312	0.01304	0.01348
Sm^{149}		0.00666	0.00663	0.00660	0.00648	0.00630	0.00618	0.00620
Gross fission products		0.04320	0.04893	0.05909	0.06637	0.07039	0.06733	0.06539
Leakage	0.02028	0.02035	0.02052	0.02056	0.02060	0.02053	0.02050	0.02058
Total	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

Table 11.18. Neutron Balances for Typical Cycles for HTGR Case 10

	Initial Balance, k = 1.005	Balance at End of Cycle						
		1	2	4	6	10	20	55
Th^{232}	0.36640	0.36758	0.36413	0.35430	0.34875	0.34603	0.34611	0.35672
Pa^{233}		0.00864	0.00846	0.00843	0.00828	0.00810	0.00808	0.00855
U^{233}		0.17296	0.18260	0.22358	0.25989	0.29230	0.32075	0.33259
U^{234}	0.00236	0.00654	0.00732	0.01028	0.01352	0.01908	0.02857	0.03014
U^{235}	0.47634	0.27220	0.26079	0.21281	0.17163	0.13490	0.10401	0.09140
U^{236}	0.00019	0.00746	0.00785	0.00951	0.01080	0.01237	0.00991	0.00368
U^{238}	0.00411	0.00381	0.00377	0.00368	0.00360	0.00363	0.00271	0.00120
Pu^{238}		0.00017	0.00025	0.00058	0.00091	0.00124	0.00168	0.00037
Pu^{239}		0.00390	0.00393	0.00388	0.00372	0.00349	0.00269	0.00114
Pu^{240}		0.00148	0.00147	0.00149	0.00140	0.00129	0.00103	0.00042
Pu^{241}		0.00104	0.00113	0.00128	0.00125	0.00108	0.00095	0.00036
Pu^{242}		0.00003	0.00000	0.00001	0.00001	0.00009	0.00007	0.00003
Be	0.06723	0.06746	0.06779	0.06814	0.06826	0.06828	0.06808	0.06870
B	0.05593							
C	0.00443	0.00460	0.00470	0.00487	0.00495	0.00485	0.00484	0.00519
O	0.00272	0.00272	0.00273	0.00274	0.00274	0.00274	0.00274	0.00274
Xe^{135}		0.01360	0.01368	0.01377	0.01374	0.01355	0.01345	0.01390
Sm^{149}		0.00655	0.00651	0.00657	0.00640	0.00613	0.00598	0.00603
Gross fission products		0.03890	0.04240	0.05349	0.05953	0.06028	0.05786	0.05623
Leakage	0.02030	0.02036	0.02049	0.02059	0.02062	0.02057	0.02050	0.02061
Total	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

Table 11.19. Mass Balance Over Life of HTGR

Case	1	8	9	2	3	4	10	5	B-10	B-5	B-55	A-4	A-10	A-5	AB-5	AC-10
Initial k																
First cycle	1.0925	1.0925	1.0925	1.0925	1.0925	1.08	1.086	1.0925	1.086	1.0925	1.10	1.08	1.086	1.0925	1.0925	1.10
Last cycle	1.0467	1.04	1.04335	1.0467	1.0467	1.04	1.0433	1.0467	1.0433	1.0467	1.05	1.04	1.0433	1.0467	1.0467	1.05
History time, years	30.6	30.3	30.2	30.2	30.7	28.2	30.4	30.6	28.3			29.1	30.6	30.0	30	20.6
Last cycle time, days	265	186	202	198	236	101	155	198	104	151	192	103	153	199	151	71.7
Initial loading, kg																
U^{234}	39.6	39.7	39.7	32.3	32.3	37.5	38.5	39.7	31.2	32.1	33.1	37.5	38.5	39.7	32.3	28.9
U^{235}	3569.3	3569.3	3569.3	3558.6	3558.6	3375.8	3466.4	3569.3	2808.1	2884.2	2976.4	3375.8	3466.4	3569.3	2909.8	2595.5
U^{236}	8.5	8.5	8.5	8.4	8.4	8.0	8.2	8.5	6.7	6.8	7.1	8.0	8.2	8.4	6.9	6.2
U^{238}	216.9	216.9	216.9	216.3	216.3	205.2	210.7	216.9	170.7	175.3	180.9	205.1	210.7	216.9	176.9	157.8
Th^{232}	95,724	95,724	95,724	95,724	95,724	95,724	95,724	95,724	83,759	83,759	83,759	95,722	95,724	95,724	83,759	54,388
Total feed material (excludes that recycled), kg																
U^{234}	135.0	108.7	121.0		91.60	93.8	92.2	99.2	31.21			83.7	89.3	81.8	76.36	87.1
U^{235}	11,066	8,406	9,561		9,607	8,439	8,294	8,925	7,301			7,537	8,038	7,359	6,870	7,837
U^{236}	27.0	20.5	23.3		23.5	20.6	19.6	21.0	17.3			17.9	19	17.4	16.3	18.6
U^{238}	845.3	713.9	788.0		583.6	512.9	504.1	542.4	443.8			458.1	488.5	447.3	417.6	476.1
Th^{232}	120,403	125,131	122,646		121,861	126,900	124,600	122,600	113,000			127,700	122,900	124,600	111,330	73,732
Total material removed, kg																
U^{233}	8,739	12,307	10,017		8,742	12,836	10,294	9,038	11,234			12,649	8,812	9,794	8,993	8058.7
U^{234}	2,410	3,488	2,826		2,521	2,702	2,539	2,330	2,627			2,662	2,241	2,394	2,351	1776.8
U^{235}	3,632	3,996	3,523		2,852	3,501	2,847	2,689	2,690			3,065	2,298	2,369	1,972	3022.4
U^{236}	3,351	3,958	3,580		1,748	1,714	1,815	1,930	1,528			1,615	1,812	1,690	1,556	1458.8
U^{238}	903.8	1,054	948.1		448.5	581.3	440.0	460	497.5			528.9	501.4	483.8	444.2	527.1
Th^{232}	374,087	561,610	441,375		395,459	653,100	494.000	418,100	568,400			650,100	414,900	476,000	439,440	367,374
Total material recycled, kg																
U^{233}	5,911	9,686	7,321		6,152	9,208	7,373	6,297	8,313			9,207		7,057	6,644	6,101
U^{234}	1,470	2,555	1,892		1,598	1,193	1,210	1,136	1,210			1,204		1,166	1,172	782
U^{235}	1,403	2,124	1,650		547.4	1,250	1,186	1,097	1,018			1,153		1,018	815	984
U^{236}	1,299	2,188	1,603		195.3	450.3	492.9	526.7	383.4			466.5		513.6	436.3	332.2
U^{238}	342.3	561.2	412.8			171.9	171.2	170.6	144.5			169.4		166.0	139.8	135.1
Th^{232}	272,544	456,748	338,069		293,611	546,700	390,100	315,400	475,600			543,800		372,300	348,642	307,100
Last cycle feed (excluding recycle), kg																
U^{234}	3.2	1.0	1.7	1.2	1.5	0.8	1.0	1.5	0.70	0.97	1.4	0.67	0.49	1.2	0.84	0.83
U^{235}	284.9	87.0	151.3	132.5	170.1	73.1	92.3	131.4	63.2	87.2	124.0	60.1	43.7	108.5	75.5	74.56
U^{236}	0.7	0.2	0.4	0.3	0.4	0.2	0.2	0.3	0.15	0.21	0.4	0.14	0.1	0.26	0.18	0.18
U^{238}	17.3	5.3	9.2	8.1	10.3	4.4	5.6	8.0	3.8	5.3	7.5	3.7	2.7	6.6	4.6	4.53
Th^{232}	554.9	399.0	466.0	507.6	115.7	325.7	436.7	488.9	318.3	405.1	481.0	334.7	433.2	503.7	401.3	204
Last cycle recycle feed, kg																
U^{233}	191.3	177.3	178.5	155.1	175.8	126.9	146.2	153.8	115.9	128.2	136.2	126.3	141.8	149.9	124.5	83.2
U^{234}	62.3	62.4	57.2	30.3	60.8	16.4	25.2	29.0	17.1	23.9	29.2	16.4	23.7	28.5	22.5	10.4
U^{235}	55.6	45.6	43.6	7.6	25.8	2.7	5.5	7.0	3.0	5.2	7.3	2.7	5.1	7.0	4.8	1.55
U^{236}	83.6	62.9	67.4	1.2	13.7	0.2	0.7	1.0	0.31	0.7	1.3	0.26	0.66	1.1	0.69	0.13
U^{238}	19.3	13.9	16.2	0	0	0	0	0	0	0	0	0	0	0	0	0
Th^{232}	6808.4	6964.4	6897.4	6855.8	7247.7	7037.7	6926.5	6874.5	6124.7	6037.9	5962.0	7028.7	6930.1	6859.6	6041.7	3979
Last cycle one zone discharge (total to reprocessing), kg																
U^{233}	196.6	184.6	182.7	168.2	179.1	153.2	163.1	168.1	134.3	140.3	145.1	149.8	157.2	161.6	135.6	100.8
U^{234}	64.7	63.3	59.9	51.1	61.7	35.2	46.7	51.5	34.8	43.2	48.1	34.4	43.9	49.0	40.3	22.7
U^{235}	58.5	47.3	46.2	32.0	40.2	26.4	28.0	27.5	20.6	19.3	25.7	22.3	24.4	24.6	20.6	23.6
U^{236}	85.5	39.6	67.2	25.5	35.4	12.0	18.2	18.7	11.3	12.5	24.2	10.3	15.6	17.9	15.4	11.2
U^{238}	19.2	14.8	15.8	5.8	6.13	3.7	4.4	3.8	3.2	2.5	5.2	3.1	3.7	3.5	3.6	3.9
Th^{232}	6865.8	7032.2	6959.0	6931.9	6874.7	7108.8	6996.4	6943.4	6186.6	6093.2	6021.3	7099.7	7000.2	6916.2	6103.3	4019.9
Last cycle one zone discharge (to reprocessing for recycle), kg																
U^{233}	196.6	184.6	182.7	156.4	179.1	128.2	147.7	156.3	117.1	129.9	137.8	127.6	143.2	152.3	125.7	84.0
U^{234}	64.7	63.3	59.9	30.1	61.1	16.5	25.5	29.9	17.3	24.5	29.5	16.6	23.9	29.6	22.7	10.5
U^{235}	58.5	47.3	46.2	7.4	26.2	2.7	5.5	7.3	3.0	5.4	7.4	2.8	5.1	7.4	4.8	1.6
U^{236}	85.5	39.6	67.2	1.1	13.7	0.2	0.7	1.1	0.32	0.78	1.3	0.26	0.67	1.2	0.70	0.13
U^{238}	19.2	14.8	15.8	0	0	0	0	0	0	0	0	0	0	0	0	0
Th^{232}	6865.8	7032.2	6959.0	6931.0	6874.7	7108.8	6996.4	6935.1	6186.6	6093.2	6021.3	7099.7	7000.2	6916.2	6103.3	4019.9

Table 11.20. Comparison of Reference Design SGR with System Calculated by AI

Item	Reference Design Presented to ORNL	System on Which AI Results Are Based
Central fuel element support tube	Stainless steel, 0.570 in. OD, 0.030-in. wall	Zircaloy, 0.570 in. OD, 0.060-in. wall
Number of fueled lattice positions	594[a]	592
Number of control rod lattice positions	102[a]	84
Control rod cooling	"Wet rods," cooled with sodium	"Dry rods," no sodium in rod channels
Voids in graphite	~4%	~2.5%

[a]Revised from report NAA-SR-9213 (ref. 19, Chap. 3) at the suggestion of Atomics International.

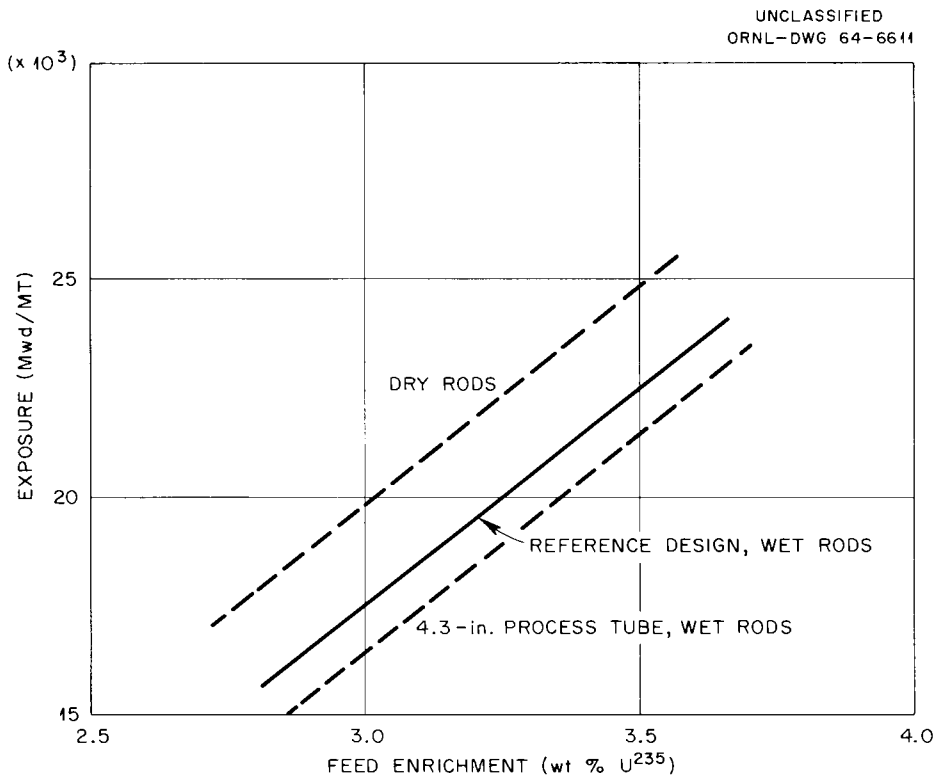

Fig. 11.10. Equilibrium-Cycle Exposures for SGR Versus Feed Enrichment.

ORNL results were based on the 4.3-in.-OD process tube specified in the reference report (changed to 4.0 in. by AI), results are also shown for the larger tube. Mass balances are presented in Tables 11.21 and 11.22.

Results from AI indicate that they would predict that an exposure of 35,000 Mwd/MT could be achieved with 4.1% enrichment with dry control rods. Extrapolating the upper line on Fig. 11.10 yields 4.5% from ORNL calculations for a dry rod design. For the reference design, which has wet rods, the feed enrichment would be 4.77% at 35,000 Mwd/MT.

It appears that ORNL calculations are probably somewhat less optimistic than calculations by AI, even for the same reactor design, although a direct comparison of results for exactly the same case was not made. However, a significant difference in the results being compared is caused by the differences in reactor description discussed above.

Summary

The physics analyses performed in this evaluation treated a complete 30-year history of reactor operation. This method of analysis compared the reactors in a more realistic manner than the conventional equilibrium cycle analysis. However, optimization of the 30-year history is a much more formidable task than optimizing the equilibrium cycle.

Physics calculations were made with a multigroup point depletion code in order to allow parameter studies to be made with a resonable computing time. With the large cores studied in this evaluation, this model should be adequate to provide a reliable comparison of the concepts.

There remains an uncertainty in the epithermal cross sections for U^{233}. Cross sections based on fast chopper measurements yield a lower average epithermal eta value than do integral measurements. Because of this unresolved discrepancy in the data, calculations were made using both U^{233} cross section sets. The higher eta value appears to be more realistic, based on recent measurements at KAPL. Therefore, most of the comparisons in this study were based on the higher value.

Table 11.23 presents the feed enrichment, exposure, and conversion ratio for the minimum fuel cost case from these studies. Table 11.24 presents the fuel makeup requirements for the entire 30-year operating life of the plant. In the last two columns of Table 11.24, the makeup requirements are expressed in terms of the consumption of ore reserves.

The results from these calculations differ from the results obtained by the sponsors to varying degrees. It is not surprising that there would be such differences between any two independent calculations, since calculational techniques and cross section libraries are far from standardized. The comparison of ORNL results with those of the sponsor are discussed briefly below.

1. PWR. ORNL results for the PWR are more optimistic than the Westinghouse results presented in the reference report, but they are in good agreement with studies made by MIT and Combustion Engineering.

2. SSCR. The results for the SSCR are in substantial agreement with those of Babcock & Wilcox. Conversion ratios calculated by ORNL are a few points lower than Babcock & Wilcox values. Also, for a given enrichment, the initial D_2O ratios calculated by ORNL are higher than

Table 11.21. Summary of Cycle Performance for SGR

	Reference Design Cases			Dry Rod Cases	
	1A-1	1A-2	1A-3	3A-2	3A-4
First cycle					
Initial k (no rods)	1.110	1.115	1.120	1.115	1.125
Cycle time, full-power days	186	194	202	184.6	200.0
Cycle exposure, Mwd/MT of U	6257	6523	6790	6223	6738
Initial enrichment, wt % U^{235}	2.396	2.433	2.472	2.178	2.247
Equilibrium cycle					
Initial k (no rods)	1.060	1.065	1.070	1.065	1.075
Cycle time, full-power days	81.4	95.0	110.3	89.00	119.3
Discharge exposure, Mwd/MT of U	16,350	19,030	22,022	17,888	23,829
Feed enrichment, wt % U^{235}	2.867	3.138	3.450	2.803	3.408
Discharge enrichment, wt % U^{235}	1.182	1.231	1.30	1.065	1.203
Discharge Pu, g/kg of U feed	5.33	5.82	6.34	5.56	6.55
Pu composition, wt %					
Pu^{239}	53.18	51.45	50.12	50.05	47.73
Pu^{240}	29.78	29.55	29.06	30.79	29.57
Pu^{241}	13.83	15.12	16.29	15.05	17.21
Pu^{242}	3.21	3.88	4.53	4.11	5.49

Table 11.22. Mass Balance Over Life of SGR

	Reference Design Cases			Dry Rod Cases	
	1A-1	1A-2	1A-3	3A-2	3A-4
History time, years	30.20	30.26	30.13	30.20	30.07
Initial loading, kg					
U^{235}	1673.9	1700.6	1728.1	1518.0	1567.5
U^{238}	68,186	68.186	68,186	68,186	68,186
Total feed material, kg					
U^{235}	36,089	34,194	32,658	32,404	30,140
U^{238}	1,272,812	1,102,326	954,563	1,170,520	886,358
Total material removed, kg					
U^{235}	15,921	14,394	13,171	13,199	11,266
U^{236}	3988	4074	4121	3828	3913
U^{238}	1,259,100	1,088,850	941,421	1,156,576	873,084
Total Pu	6826	6436	6049	6526	5778
Fissile Pu	4592	4312	4046	4276	3781
Equilibrium cycle feed, kg					
U^{235}	335.4	368.1	406.1	327.7	401.0
U^{238}	11,364	11,364	11,364	11,364	11,364
Equilibrium cycle discharge, kg					
U^{235}	134.8	140.3	148.1	121.2	136.8
U^{236}	32.1	38.6	46.5	34.1	49.7
U^{238}	11,238	11,220	11,200	11,224	11,184
Total Pu	62.3	68.3	74.6	65.1	77.1
Fissile Pu	41.8	45.5	49.6	42.4	50.0

Table 11.23. Conversion Ratio Results for Near-Optimum Fuel-Cycle Cost Based on ORNL Physics

Concept	Fuel Management	Case No.	Feed Enrichment[a]	Exposure (Mwd/MT of U + Th)	Conversion Ratio[b]	(1.0 − CR)/EFF, Index of Fuel Consumption[c]
PWR-U	Three zone[d]	7	2.2	21,050	0.60	1.29
SSCR-Th	Single batch	B-3	4.0	21,100	0.75	0.80
HWR-U	Three zone[d]	E-1	1.1	11,100	0.66	1.27
HWR-Th	On-power fueling	G-4	2.1	28,800	0.84	0.60
HTGR-Th	12 zone[d]	AB-5	2.1	52,300	0.90	0.22
SGR-U	Six zone[d]	IA-2	3.1	19,000	0.47	1.22

[a]Percent by weight based on total heavy metal.

[b]Gross estimated average for last cycle; high eta U^{233} results considered, except for SBR, for which calculations were not made.

[c]A measure of relative fuel consumption (last cycle), CR = average conversion ratio, EFF = electrical energy ÷ thermal energy; does not account for differences in fuels.

[d]Number of individual loadings per core.

those of Babcock & Wilcox. Recent changes in the Doppler-broadened thorium cross sections used by both ORNL and Babcock & Wilcox should produce closer agreement in results.

3. HWR-U. For 1.2% feed fuel enrichment, the HWR-U exposure calculated by ORNL was 12,540 Mwd/MT compared with 15,400 Mwd/MT calculated by Du Pont. This difference has not been accounted for at this time.

4. HWR-Th. ORNL and Du Pont results appear to be in good agreement, although the physics results have not been completely compared in all details.

5. HTGR. Based on the limited comparison possible, it appears that the conversion ratios calculated by ORNL for the HTGR are 0.02 to 0.05 lower than those calculated by GA. The source of this difference has not been determined, but it is probably the combined effect of a number of small differences.

6. SGR. ORNL results for the SGR are quite pessimistic in comparison with AI results. A significant part of the difference is due to differences between the reference design submitted to ORNL and that calculated by AI. However, there appear to be other differences in the basic physics that have not been fully resolved at this time.

Although the ORNL physics results were not in exact agreement with those of the sponsors, the results of this study should provide a valid basis for comparison of the concepts. All the calculations were made with the same computer code using a consistent set of nuclear data.

Where there were significant differences between ORNL and the sponsor's results, these differences were acknowledged by giving an appropriate range of fuel cycle costs in Chapter 12.

Table 11.24. Summary of Fuel Consumption for Each Concept Estimated from ORNL Physics

Near-Optimum Fuel-Cycle-Cost Mass Balances Considered

Reactor Concept	Case	Initial Fuel Loading for 1000 Mw(e) (kg of U^{235})	Thirty-Year History Fuel Balance			Final Fuel Loading (kg of fissile material)	Thirty-Year Ore Consumption[a] [g of U + Th[b] per Mwhr(e)]	
			Total Feed to Develop 800 Mw(e)[c] (kg of U^{235})	Total Discharge (kg of fissile material)	Net Consumption[d] (kg of fissile material)		Total Feed Basis (no credit for discharges)	Net Consumption Basis
PWR-U	7	2060	30,600	14,700	18,800	1400	28.2	17.3
SSCR	B-3	3340	14,600	2,900	11,800	2700	15.6	12.4
HWR-U	E-1	580	33,200	17,300	23,600	490	26.7	19.0
HWR-Th	G-4	1440	9,440	1,760	7,710	1670	10.4	8.2
HTGR-Th, with BeO	AB-5	2910	6,870	3,506	3,413	1970	7.6	3.6
HTGR-Th, no BeO[e]	AC-10	2600	10,700	5,200	5,600	1680	11.5	5.9
SGR	1A-2	1700	33,900	18,500	17,400	1600	32.4	16.6

[a]Considering reenrichment of discharged uranium; fissile plutonium credited against consumption.

[b]Thorium consumption is a small fraction with recycle and recovery.

[c]Excluding recycle but including initial loading.

[d]Including allowance for fabrication and processing losses.

[e]Single no-BeO case calculated; probably not representative due to low exposure.

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12. FUEL-CYCLE COSTS

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The costs incurred in the procurement, fabrication, shipping, irradiation, reprocessing, and waste disposal of nuclear fuel comprise the fuel-cycle cost. The cost is directly proportional to the charges for these services but is indirectly proportional to the amount of energy released by the fuel and the thermodynamic efficiency of the energy conversion system. The fuel-cycle cost is a very significant criterion in the evaluation of reactor concepts because it is a measure of the recurring charges in reactor operation. In fact, in some cases, for example, different reactors of approximately equal capital and power conversion investment, the fuel-cycle cost may be the only economic guide in comparing the systems. Furthermore the importance of fuel-cycle cost as a component of the overall power cost is influenced by the financial structure that is applied to the reactor and power conversion plant. More favorable financing of the plant tends to put more emphasis on fuel-cycle cost.

For clarity and convenience, fuel-cycle cost is divided into eight component costs: fabrication, burnup and losses, reprocessing and conversion, shipping, fissile and fertile inventory, interest on fabrication, interest on reprocessing, and interest on shipping. A ninth cost component that is unique to some reactors is the D_2O inventory charge; this charge is listed separately in the power cost tabulation.

Fabrication Cost

The fabrication cost includes all charges for the manufacture of a finished fuel element that is ready for insertion into a reactor. The fabrication process begins with fresh or recycle fertile and fissile oxides, ThO_2 or UO_2 or both, and includes sizing, blending, screening, filling, compacting, inspection, and assembly operations to produce a finished oxide fuel element. In the case of carbide fuels, carbiding and pyrolytic-carbon coating (HTGR only) operations are part of the cycle. The fabrication cost also includes the cost of sol-gel operations for the preparation of oxides from the $Th(NO_3)_4$ and $UO_2(NO_3)_2$ product of the chemical reprocessing plant. For nonrecycle fuels the sol-gel operations are carried out on purchased oxides.

The fabrication cost is composed of three principal parts: fixed charges on the capital investment in a plant, daily operating charges, and hardware cost. Fixed charges were calculated at 15, 22, and 30% per year, with a reference value of 22% per year being used for comparison of results. Hardware cost includes the charges for all components of the fuel element that are necessary to contain the fuel and fertile materials and to make an operable element. This cost is always a significant portion of the overall fabrication cost and, in some cases, is the major cost component. This is brought about by the expensiveness of nuclear-grade materials (Zircaloy, BeO, and graphite) and the large quantity of careful work required to satisfy the intricate design and close tolerances of the component parts of

an element. Fabrication plant operating charges cover the cost of labor, services, and materials required for day-by-day operation.

Sol-gel operations are carried out at the chemical reprocessing plant because the process makes a solid oxide product that is readily shipped and stored. The charges, however, are included in the fabrication cost because oxide production is logically the first step in fabrication and must be used whether or not the fuel is recycled.

Burnup and Losses

The burnup charge is assessed on both fissile and fertile materials and is computed as the difference in value of these materials entering the reactor and their value at discharge. Fertile burnup is generally negligible, since it represents only the net change in fertile material from neutron absorptions. Allowance at annual rates of 1 and 0.2%, respectively, is made for nonrecoverable losses in both the reprocessing and fabrication plants. In the case of a recycle system the burnup and loss charge is the amount of money that has to be spent in purchasing material from an outside source in order to sustain reactor operation.

Processing and Reconversion Costs

The processing cost includes all charges for mechanical and chemical operations on the irradiated fuel required for adequate decontamination from fission products. Thorium-uranium fuels are considered to be processed using the Acid-Thorex flowsheet, and natural and low-enrichment uranium fuels by the Purex process. The three major components of cost are the amortization of the capital investment, operating charge, and ultimate waste disposal. Amortization rates of 15, 22, and 30% per year have been applied to the capital investment, with a reference value of 22% per year being used for comparison. The operating charge represents the cost of labor, services, and materials required for daily operation. The cost of ultimate waste disposal is associated with calcining, transporting, and storing fission products in a salt mine. Interim waste storage is provided by tankage at the processing site.

Reconversion costs are applied to nonrecycle uranium that must return to the gaseous diffusion plant for reenrichment. The cost is that of converting $UO_2(NO_3)_2$ to UF_6. The USAEC base charge of \$5.60 per kg of uranium was made for the conversion of material that contained 5 wt % or less U^{235}. No conversion charge is applied to material containing U^{233}, since it is acceptable by the AEC in nitrate solution.

No conversion charge is made for plutonium, since the reference value established in the ground rules (Chap. 2) is for nitrate solution.

Shipping Cost

The shipping cost includes the amortization of the capital investment in shipping casks, transportation between sites by rail freight, cask handling and decontamination expense, and property loss insurance. For the recycle fuels, these costs are incurred in shipping recycle fuel from the fabrication plant to the reactor and in shipping irradiated fuel from the reactor to the processing site. There is no shipping between the processing and fabrication plants, since they are on the same site. Each shipment is made in a cask that is especially designed for the radiation protection and heat-removal requirements of the fuel at the time of shipment, and it is assumed that shipping casks can be shared among reactors of the same type.

Natural and slightly enriched uranium fuels require an additional shipment to the two trips mentioned above. These fuels are not recycled, but after processing they must be returned to a gaseous diffusion plant for reenrichment. The decontaminated fuel, $UO_2(NO_3)_2$, is first converted to UF_6 and then shipped in cylinders. A nominal charge of \$1/kg of uranium was applied to this shipment. This value was assumed to cover cask rental, transportation, handling, and insurance.

The only insurance that is considered in the shipping cost is property loss insurance. The cost of insurance for each shipment was taken to be 0.4% of the value of the shipment.

Fissile and Fertile Inventory

This charge represents the interest that is paid on money invested in fissile and fertile material in the reactor system and external to the reactor system. In determining the external holdup time, it was assumed that recycle fuel could be shared among reactors of the same type and that the discharge time intervals were evenly spaced among the several reactors. These assumptions allowed the calculation of a maximum external holdup time that should be charged to a fuel batch from a given reactor. Preirradiation holdup time was determined by summing the prefabrication, fabrication, and shipping times and a 60-day preexposure period at the reactor. In calculating the postexposure holdup, an economic balance was made between the shipping cost, processing cost, and the fuel inventory charge in order to obtain the economic optimum holdup time.

Holdup times for selected values of fuel exposure are given in Table 12.1. The last column shows the frequency with which a batch of fuel is being received at the processing plant, processed, and fabricated. This time is the maximum postfabrication delay time that a reactor should have before its recycle fuel can be sold to another reactor that is being refueled.

The interest charge on the fissile and fertile inventory was computed as a percentage of the amount of capital tied up in these materials. The amount of capital changes with the purchase and sale of fuel and with the receipt of power income. For tax purposes, fuel was taken as depreciating linearly from its initial to its final value during an irradiation period.

Table 12.1. Typical Preirradiation, Irradiation, and Postirradiation Holdup Times

Installed Capacity: 15,000 Mw(e)

Reactor	Batches per Reactor	Pre-exposure Holdup (days)	Post-exposure Holdup (days)	Interval Between Discharge of Batches from Reactors (days)	Maximum External Holdup (days)
PWR	3	113	220	24	333[a]
SSCR	1	163	186	51	400
HWR-U	3	96	282	8	378[a]
HWR-Th	(b)	98	229	6[c]	333
HTGR	12	113	217	24	354
SGR	6	104	165	15	269[a]

[a]Fuel not recycled; total holdup does not include interval between discharges.

[b]Continuously fueled.

[c]For processing economy, fuel is batched through plant on six-day cycle.

A reference value of 10% per year, which includes the cost of money, income tax, and local taxes, was used for the interest charge. The effect of using 6 and 12% was also computed for typical cases.

Interest on Fabrication Working Capital

Interest at the rate of 10% per year was charged on money spent to fabricate fuel. Interest is paid on the unpaid balance of this debt from the time the fuel is received at the reactor plant until its discharge from the reactor. The debt is decreased linearly with time during irradiation through the sale of power, and when irradiation has been completed, the full fabrication value of the fuel has been recovered and the debt has been eliminated.

Processing Interest

A batch of fuel does not require processing until after revenue has been received for the power which it produced. This revenue is used to reduce existing debt or for investment at the currently prevailing interest rate until it is needed to pay the cost of processing the batch. In either case a net savings is realized, and, consequently, processing interest may

be a negative quantity. The rate of return on money set aside to pay processing charges was taken as 6% per year. Use of the present-worth method of accounting is directly equivalent to the computation of compound interest.

Shipping Interest

The portion of the total shipping cost that is incurred prior to irradiation is paid for upon delivery of the fuel to the reactor with money that has been borrowed. The interest rate on the unpaid balance of this debt is taken as 6% per year until liquidation of the debt during the irradiation period for the particular batch of fuel. On the other hand, postirradiation shipping costs can be anticipated by setting aside funds at the prevailing rate of return (6% per year) so that there is credit received rather than interest charged. Since the postirradiation shipping cost is significantly greater than the corresponding preirradiation cost, the net shipping interest charge is generally a credit rather than an expense.

Method of Computation

In determining the fuel-cycle cost for a particular reactor, its operating history was followed, cycle by cycle, over its lifetime. In the case of the thorium-uranium reactors, startup was made using unirradiated Th-U^{235} fuel; subsequent refuelings of the batch-cycle SSCR were made with recycle fuel, but continued additions of unirradiated fuel were made to the HWR-Th and HTGR until adequate time for processing and refabrication was past. Further fuel additions were recycle fuel plus fresh Th-U^{235} as required to satisfy burnup and losses. The reactors that use slightly enriched uranium were treated in the same way, except that no fuel was recycled.

The fuel-cycle cost was determined by calculating the present worth of expenditures for fuel purchase, fabrication, reprocessing, and shipping over the 30-year reactor lifetime and dividing these expenditures by the present worth of the power produced during the same period. This cost was broken down into categories by the following procedure. The direct cost (net) of expenses and credits in each category was divided by the total power produced during the lifetime of the reactor. The values obtained, expressed in mills/kwhr, are reported as "fabrication cost," "burnup and losses," "processing cost," and "shipping cost." Using the present-worth procedure, the net present value of all expenses and credits in each of the four categories was then computed and divided by the present value of the power produced.

The direct costs were subtracted from these "total" costs, and the differences termed "uranium inventory," "interest on fabrication," "interest on processing," and "interest on shipping." Using a 6% discount factor, the interest rate on these items is effectively 6%. Because fuel fabrication costs and fuel purchases are depreciated for tax purposes over the period of use rather than deducted immediately as an

operating cost, the applicable interest rate on fabrication cost and fuel inventory is 10% (in the reference case). Allowance is made for this by multiplying the "uranium inventory" and "interest on fabrication" values by the ratio of 10 over 6. Where plutonium is produced, the credit for its sale and an inventory charge are computed in a similar manner.

In effect, the direct costs computed as described above are those that would occur if the interest rate were zero, and the inventory and interest values are the additional costs (or credits) that occur because of the cost of money. The present-worth procedure followed in this study is derived and explained in greater detail in Appendix F.

Chronology of Fuel Cycle Events

A reactor fuel cycle can be divided in several distinct events which occur in time as shown in Fig. 12.1. The diagram assists in identifying the points in time at which events occur and at which charges are made or credits granted in the present-value accounting procedure. If the reference zero time is taken as the beginning of fuel exposure in the reactor, it will be noticed that certain expenditures occur prior to receipt of revenue, for example, at point (1) when a new batch or makeup fuel is purchased, and at point (2) when the fabrication and shipping charges are paid. On the other hand, revenue is received for power produced by the fuel before the irradiated fuel shipping cost is paid (point 3) and before the processing cost is paid (point 4).

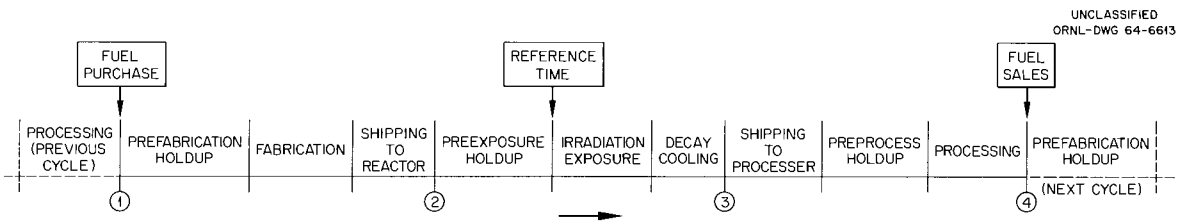

Fig. 12.1. Schematic Representation of Time Steps in a Fuel Cycle.

It is assumed that the receipt of revenue from the sale of power occurs semiannually, beginning six months after initial startup of the reactor. The last revenue is received six months after the reactor is shut down at the end of its 30-year life.

End-of-Life Value of Fuel

At the end of the first integral fuel cycle after 30 calendar years of operation the reactor is completely discharged and the fuel is sold. In all cores, except single-batch cores, there is unused fabrication

value in the fuel, and it is assumed that this fuel can be further irradiated in the same or another reactor without processing. Credit for the remaining fabrication value is granted at the end of the last cycle, but a partial processing charge is made at the same time; both are computed for each batch from the elapsed fraction of full-cycle time. Completely exposed fuel is processed and sold.

Unit Costs Used in Calculating Fuel-Cycle Costs

In determining the unit costs to be used in calculating the fuel-cycle costs for these reactors, the following procedure was used. The fuel exposure was based on the equilibrium cycle in the nonrecycle systems and on a cycle late in the 30-year history for the recycle systems (which do not reach a true equilibrium). With this fuel exposure level, the plant capacities for the reference 15,000-Mw(e) industry were determined and the unit costs for fabrication, reprocessing, and shipping were taken from Chapters 8, 9, and 10. Since there are many curves presented in Chapter 8, the specific sources used in the fuel-cycle cost calculations are listed below:

Reactor	Fuel-Cycle Cost Source
PWR	Fig. 8.1
SSCR	Fig. 8.6
HWR-U	Fig. 8.7
HWR-Th	Fig. 8.10
HTGR	Figs. 8.16 and 8.18
SGR	Fig. 8.19.

Fuel-Cycle Costs for PWR

Fuel-cycle costs were calculated for the PWR based on the physics results presented in Chapter 11. Costs averaged over 30 years of reactor operation are presented in detail in Table 12.2. Shown in Fig. 12.2 is the dependence of fuel-cycle cost on feed enrichment and the sale value of the fissile plutonium.

As noted in Chapter 11, the ORNL physics results are somewhat optimistic relative to Westinghouse results. The effect of this difference on fuel costs was estimated by applying ORNL cost bases to Westinghouse physics results (equilibrium Zircaloy-clad core from Table VI-4 of the reference report). Costs for equilibrium cycles are compared in Table 12.3.

The minimum fuel cost calculated with ORNL physics results was 1.67 mills/kwhr. Comparing equilibrium cycle results, it appears that the cost would be about 0.10 mill/kwhr higher with Westinghouse physics. Thus, the minimum fuel-cycle cost for the PWR is about 1.7 to 1.8 mills/kwhr.

Table 12.2. Thirty-Year Fuel-Cycle Costs of a 1002-Mw(e) Pressurized-Water Reactor (PWR)

Fixed charge on capital investment in fabrication and processing plants: 22% per year
Fabricating days per year: 260
Processing days per year: 194
Plutonium value: $7 per gram of fissile material
One-third of core discharged for refueling

Case	6	7	8	9	10	11
Equilibrium or last-cycle data						
Exposure, Mwd/MT of U	13,120	21,050	27,600	31,430	35,750	40,680
Fuel lifetime, full-power days	417	670	880	1002	1141	1300
Feed enrichment, wt % U^{235}	1.696	2.198	2.656	2.938	3.271	3.662
Fabrication and processing plant size for 15,000 Mw(e), MT of U per year	1073	669	510	448	394	345
Fabrication cost, $/kg of U	47.00	52.50	56.50	58.50	61.00	64.00
Processing cost, $/kg of U	22.00	31.20	38.00	41.50	46.00	50.50
Fuel shipping costs (fabricated + irradiated), $/kg of U	3.49	4.01	4.65	5.11	5.82	6.63
Preexposure fuel holdup, days	100	107	113	117	121	125
Postexposure holdup through processing, days	190	198	204	207	211	215
Initial fissile inventory, kg	1866	2060	2207	2287	2372	2462
Fuel inventory cost of first core, dollars	13.04×10^6	15.05×10^6	16.59×10^6	17.44×10^6	18.32×10^6	19.29×10^6
Fabrication cost of first core, dollars	4.86×10^6	5.44×10^6	5.86×10^6	6.07×10^6	6.34×10^6	6.65×10^6
Fuel-cycle cost, mills/kwhr(e)						
Fabrication	0.464	0.339	0.281	0.257	0.238	0.221
Burnup + losses[a]	0.989	0.995	1.012	1.027	1.032	1.066
Processing	0.217	0.202	0.189	0.182	0.179	0.174
Shipping	0.034	0.026	0.023	0.023	0.023	0.023
Uranium inventory	0.160	0.212	0.238	0.255	0.274	0.297
Interest on fabrication	0.070	0.085	0.084	0.084	0.085	0.087
Interest on processing	-0.018	-0.012	-0.013	-0.014	-0.014	-0.015
Interest on shipping	0	-0.001	0	0	0	0
Pu credit	-0.258	-0.202	-0.172	-0.160	-0.148	-0.138
Pu inventory	0.039	0.028	0.028	0.028	0.029	0.029
Net fuel-cycle cost	1.70	1.67	1.67	1.68	1.70	1.74

[a]Losses = 1% in processing + 0.2% in fabrication.

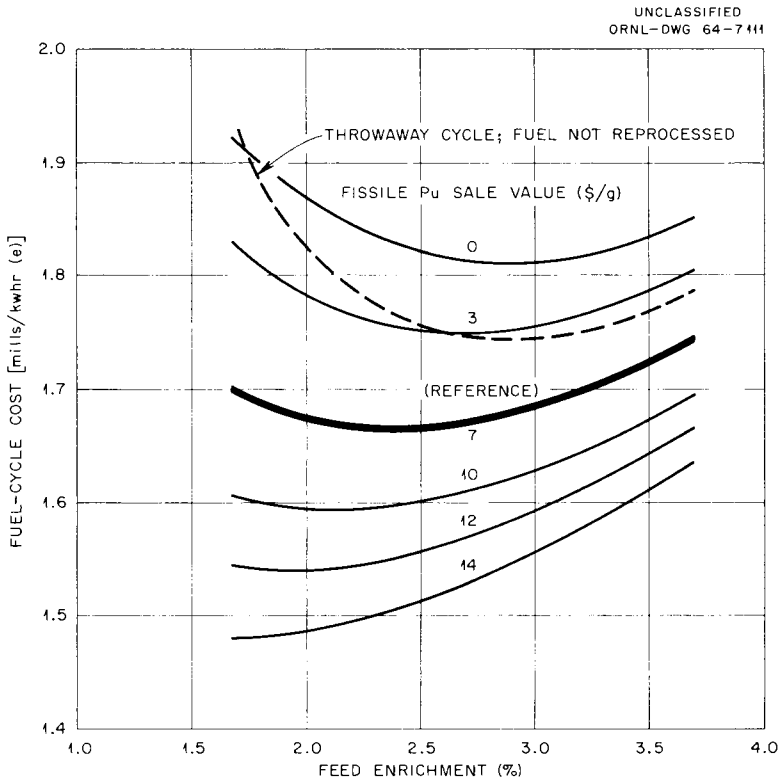

Fig. 12.2. Dependence of Fuel-Cycle Cost on Feed Enrichment and Plutonium Value for PWR.

Table 12.3. Approximate Comparison of Equilibrium-Cycle Fuel Costs Calculated for PWR from ORNL and Westinghouse Physics

	Fuel-Cycle Costs (mills/kwhr)	
	ORNL Physics[a]	WAPD Results Adjusted to ORNL Unit Costs[b]
Fabrication	0.33	0.30
Burnup and losses	0.89	1.06
Processing	0.20	0.21
Shipping	0.02	0.02
Inventory	0.22	0.28
Working capital	0.03	0.03
Plutonium credit	−0.20	−0.29
Total	1.49	1.61

[a]21,050-Mwd/MT exposure; 2.2% enrichment.

[b]24,000-Mwd/MT exposure; 3.0% enrichment.

Fuel-Cycle Costs for SSCR

Based on the physics presented in Chapter 11, fuel-cycle costs were calculated for a number of SSCR cases. Cost breakdowns are presented in Table 12.4. The C-series of cases represent results obtained with the

Table 12.4. Thirty-Year Fuel-Cycle Costs of a 1000-Mw(e) Spectral-Shift-Control Reactor (SSCR)

Fixed charge on capital investment in fabrication and processing plants: 22% per year
Fabricating days per year: 260
Processing days per year: 254
Entire core discharged for refueling

Case[a]	C-1	C-2	C-3	B-1	B-3
Equilibrium or last-cycle data					
Exposure, Mwd/MT of U + Th	38,400	22,300	30,200	34,100	22,800
Fuel lifetime, full-power days	830	460	635	720	472
Feed enrichment, wt % fissile material	7.07	4.19	5.20	5.48	4.05
Fabrication and processing plant size for 15,000 Mw(e), MT of U + Th per year	361	625	455	404	607
Fabrication cost, $/kg of U + Th	72	60	65	68	60
Processing cost, $/kg of U + Th	56	36	46.50	51	37
Fuel shipping costs (fabricated + irradiated), $/kg of U + Th	7.23	7.18	7.27	7.18	7.29
Preexposure fuel holdup, days	159	127	142	149	128
Postexposure holdup through processing, days	224	179	210	224	190
Initial fissile inventory, kg	4162	2786	3339	4163	3339
Fuel inventory cost of first core, dollars	50.7×10^6	34.2×10^6	40.8×10^6	50.8×10^6	40.9×10^6
Fabrication cost of first core, dollars	4.95×10^6	4.00×10^6	4.38×10^6	4.60×10^6	4.00×10^6
Fuel-cycle cost, mills/kwhr(e)					
Fabrication	0.233	0.339	0.275	0.237	0.282
Burnup + losses[b]	0.945	0.689	0.801	0.830	0.698
Processing	0.181	0.203	0 197	0.178	0.174
Shipping	0.022	0.039	0.029	0.024	0.033
U + Th inventory	0.893	0.692	0.780	0.779	0.625
Interest on fabrication	0.061	0.078	0.063	0.068	0.079
Interest on processing	-0.014	-0.001	-0.009	-0.013	-0.004
Interest on shipping	0	0.002	0.001	0	0.001
Net fuel-cycle cost	2.32	2.04	2.14	2.10	1.89

[a]Case numbers preceded by C are for the lower epithermal eta of U^{233} and numbers preceded by B are for the higher eta.

[b]Losses = 1% in processing + 0.2% in fabrication.

lower epithermal eta for U^{233}. Since there is an uncertainty in the U^{233} epithermal cross sections, some calculations were made with a higher epithermal eta for U^{233}. These are represented by the B-cases.

Net fuel-cycle costs averaged over a 30-year life are presented in Fig. 12.3 for both cross-section sets. Based on ORNL physics calculations, the minimum fuel-cycle cost appears to be about 1.9 mills/kwhr.

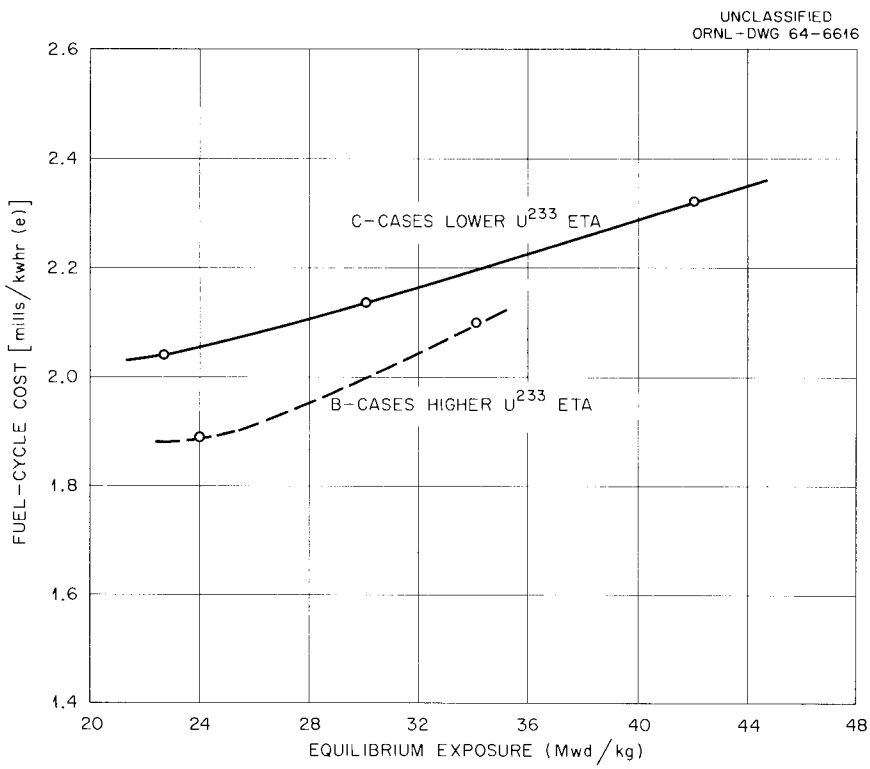

Fig. 12.3. SSCR Fuel-Cycle Cost.

Since the ORNL physics results differed somewhat from B & W results, an attempt was made to calculate fuel-cycle costs based on B & W physics and ORNL economic factors for comparison. It was not possible to obtain a direct comparison, since B & W could provide results for only a few cycles, not for a full 30-year history. The fuel cost calculated for the first cycle was 1.91 mills/kwhr and 1.65 mills/kwhr for the fourth cycle. Considering the ORNL 30-year average results and the B & W results for the few cycles mentioned above, it appears that 1.6 mills/kwhr would be about a minimum 30-year average cost based on B & W physics.

Based on the above the fuel cost for the SSCR is estimated to be in the range 1.6 to 1.9 mills/kwhr. Of this range, about 0.10 mill/kwhr is attributable to uncertainties in U^{233} cross sections, and the rest is due to other differences in physics (as discussed in Chapter 11) and the uncertain comparison of 30-year costs with equilibrium-cycle costs.

Fuel-Cycle Costs for HWR-U

Fuel-cycle costs were calculated for the uranium-fueled heavy-water reactor both on the basis of the total performance of the reactor over its 30-year history and on the basis of asymptotic performance after achievement of an equilibrium fuel cycle. Since recycle of plutonium is not considered in these calculations, equilibrium performance is reached within a very few cycles, and the results obtained by both methods are nearly the same. The fuel-cycle costs based on ORNL calculations and shown in Table 12.5 and in Fig. 12.4 have not reached a minimum for the reference condition at the lowest feed enrichment for which calculations were performed. As discussed in Chapter 11, ORNL's fuel-cycle calculations have produced slightly smaller exposures for a given enrichment than those of Du Pont. At their reference enrichment of 1.19%, the ORNL estimate of 12,540 Mwd/MT compares with that of Du Pont of 15,400 Mwd/MT for three-step partial reloading. Evaluation of fuel-cycle costs using Du Pont's equilibrium and ORNL's 30-year-life results indicates that the minimum will be below 1.2 mills/kwhr. In acknowledgment of the difference in calculated exposure, a range of 1.0 to 1.2 mills/kwhr is assigned to the fuel cost of the HWR-U.

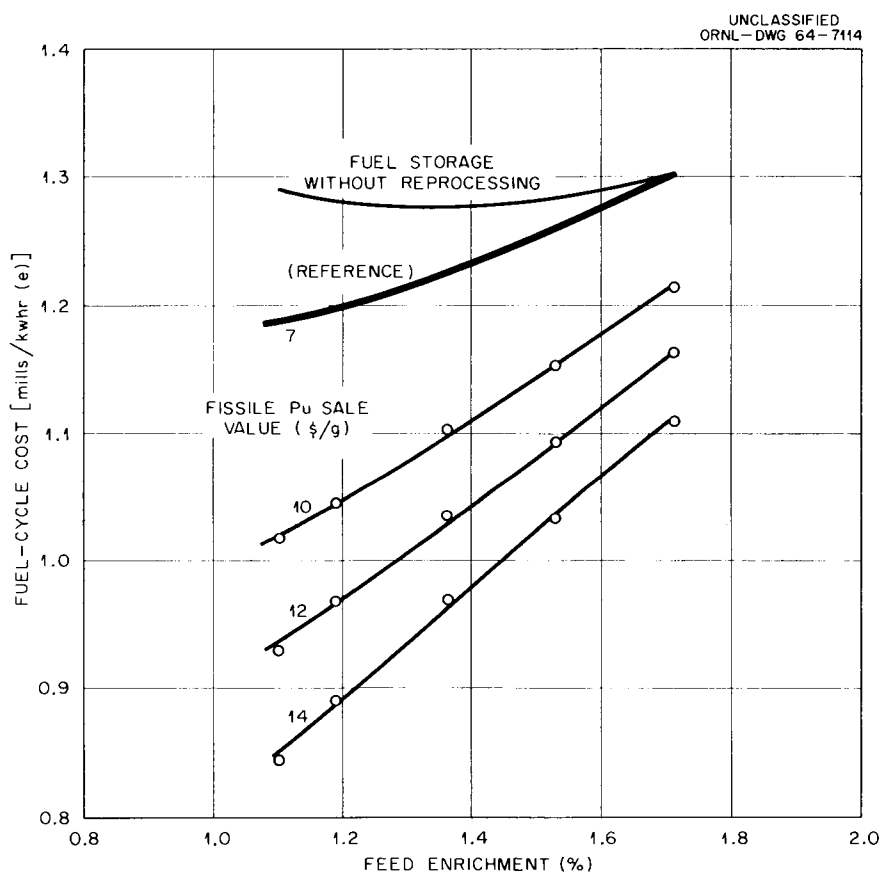

Fig. 12.4. Dependence of Fuel-Cycle Cost on Feed Enrichment and Plutonium Value for HWR-U.

Table 12.5. Thirty-Year Fuel-Cycle Costs of a 1003-Mw(e) Uranium-Loaded Heavy-Water Reactor (HWR-U)

Fixed charge on capital investment in fabrication and processing plants: 22% per year
Fabricating days per year: 260
Processing days per year: 155
Plutonium value: $7 per gram of fissile material
One-third of core discharged for refueling

Case	E-1	E-2	E-3	E-4	E-5
Equilibrium or last-cycle data					
Exposure, Mwd/MT of U	11,200	12,600	15,200	17,600	20,300
Fuel lifetime, full-power days	203	228	276	320	365
Fuel enrichment, wt % U^{235}	1.10	1.191	1.363	1.534	1.706
Fabrication and processing plant size for 15,000 Mw(e), MT of U per year	1470	1297	1076	928	804
Fabrication cost, $/kg of U	21.90	22.70	24.50	25.80	27.20
Processing cost, $/kg of U	16.50	18.20	21.00	23.80	26.80
Fuel shipping costs (fabricated + irradiated), $/kg of U	2.14	2.14	2.18	2.23	2.15
Preexposure fuel holdup, days	94	95	96	97	99
Postexposure holdup through processing, days	281	281	282	282	313
Initial fissile inventory, kg	575.7	595.6	635.5	675.3	715.2
Fuel inventory cost of first core, dollars	2.31×10^6	2.49×10^6	2.83×10^6	3.19×10^6	3.54×10^6
Fabrication cost of first core, dollars	1.50×10^6	1.71×10^6	1.68×10^6	1.76×10^6	1.86×10^6
Fuel-cycle cost, mills/kwhr(e)					
Fabrication	0.310	0.286	0.257	0.234	0.216
Burnup + losses[a]	0.808	0.817	0.851	0.882	0.914
Processing	0.234	0.229	0.220	0.215	0.213
Shipping	0.030	0.027	0.023	0.020	0.017
Uranium inventory	0.082	0.084	0.091	0.094	0.104
Interest on fabrication	0.037	0.037	0.036	0.035	0.035
Interest on processing	−0.007	−0.007	−0.007	−0.007	−0.009
Pu credit	−0.322	−0.292	−0.254	−0.227	−0.205
Pu inventory	0.016	0.017	0.017	0.016	0.018
Net fuel-cycle cost	1.19	1.20	1.23	1.26	1.30

[a]Losses = 1% in processing + 0.2% in fabrication.

In all the cases listed in Table 12.5, plutonium is recovered and sold, although at the higher enrichments the cost of reprocessing would almost offset the sale value. The discharged uranium enrichment is low enough that in no case does it have value. The calculated dependence of the fuel-cycle cost on plutonium value is shown in Fig. 12.4. At a value of $3/g of fissile material (the lower end of the range specified in the ground rules), it is cheaper not to recover the plutonium, and the upper curve represents the fuel-cycle cost for storage without reprocessing.

Fuel-Cycle Costs for HWR-Th

Fuel exposures calculated by ORNL for the heavy-water reactor operating on the thorium cycle appear to be in good agreement with those obtained by Du Pont. Evaluation of the fuel costs for the 30-year reactor life yielded values less than 0.1 mill higher than for the equilibrium fuel cycle, indicating a fairly rapid approach to equilibrium. Fuel-cycle cost details for different exposures and for both sets of epithermal eta values for U^{233} are given in Table 12.6.

As noted in Chapter 3, ORNL physics calculations did not consider a fertile blanket on this core, although one was specified in the initial

Table 12.6. Thirty-Year Fuel-Cycle Costs of a 1010-Mw(e) Thorium-Loaded Heavy-Water Reactor (HWR-Th)

Fixed charge on capital investment in fabrication and processing plants: 22% per year
Fabricating days per year: 260
Processing days per year: 155
Continuous refueling

Case[a]	A-4	A-5	G-4	G-5
Equilibrium or last-cycle data				
Exposure, Mwd/MT of U + Th	29,400	41,000	29,400	41,000
Fuel lifetime, full-power days	600	825	600	825
Feed enrichment, wt % fissile material	2.15	2.58	2.10	
Fabrication and processing plant size for 15,000 Mw(e), MT of U + Th per year	590	430	590	430
Fabrication cost, $/kg of U + Th	37.5	44.0	37.5	44.0
Processing cost, $/kg of U + Th	39.7	50.8	39.7	50.8
Fuel shipping costs (fabricated + irradiated), $/kg of U + Th	5.90	5.90	5.90	5.90
Preexposure fuel holdup, days	100	100	100	100
Postexposure holdup through processing, days	230	230	230	230
Initial fissile inventory, kg	1446	1495	1441	1490
Fuel inventory cost of first core, dollars	18.33×10^6	18.92×10^6	18.27×10^6	18.86×10^6
Fabrication cost of first core, dollars	2.99×10^6	3.50×10^6	2.99×10^6	3.50×10^6
Fuel-cycle cost, mills/kwhr(e)				
Fabrication	0.219	0.188	0.219	0.188
Burnup + losses[b]	0.419	0.512	0.411	0.493
Processing	0.242	0.226	0.242	0.226
Shipping	0.037	0.030	0.307	0.030
U + Th inventory	0.401	0.368	0.395	0.389
Interest on fabrication	0.048	0.053	0.048	0.053
Interest on processing	−0.013	−0.014	−0.013	−0.014
Net fuel-cycle cost	1.35	1.36	1.34	1.37

[a]Case numbers preceded by A are for the lower epithermal eta of U^{233} and numbers preceded by G are for the higher eta.

[b]Losses = 1% in processing + 0.2% in fabrication.

reference design. It appears unlikely that the gain in bred fuel would offset the additional costs associated with the use of a blanket.

Although continuous refueling of the reactor is provided, an assumption was made that material required for periods of operation would be purchased, received, and shipped in batches. A reduction in the inventory charge would be attained if more continuous flow of fuel were achieved.

On the basis of ORNL and Du Pont results, the 30-year average fuel-cycle cost appears to be in the range of 1.2 to 1.4 mills/kwhr.

Fuel-Cycle Cost for HTGR

Several different fueling schemes were considered for the HTGR, and these are described in Chapter 11. The study of these to determine which was optimum under the ground rules of this evaluation required computing the many cases presented in Table 12.7. Even so, insufficient cases were studied to make sure that the fuel loadings and fuel management had been completely optimized. However, certain trends appear fairly clear. In the cases studied the type I fuel management, in which the uranium is recycled only once, appears to give the best fuel-cycle costs. Figure 12.5 gives the costs as a function of fuel exposure for type I fuel management. The optimum exposure appears to be about 50,000 Mwd/MT. A range of costs is given, depending on the assumption made about the epithermal value of eta for U^{233}, and the equilibrium costs are also given for comparison.

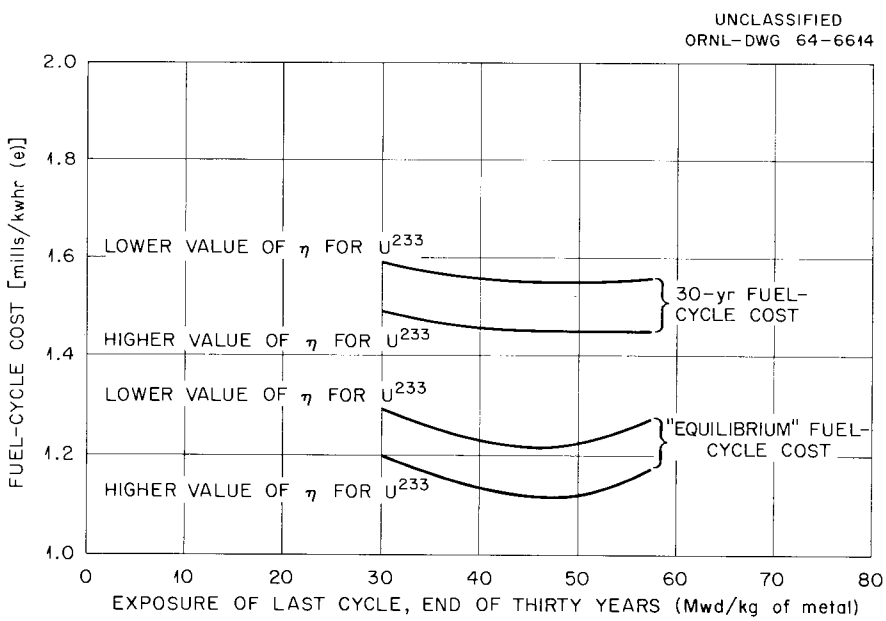

Fig. 12.5. HTGR Fuel-Cycle Costs at a Beryllium-to-Thorium Ratio of 28.

Table 12.7. Thirty-Year Fuel-Cycle Costs of a 1008-Mw(e) High-Temperature Graphite Reactor (HTGR)[a,b]

Fixed charge on capital investment in fabrication and processing plants: 22% per year
Fabricating days per year: 260
Processing days per year: 195
One-twelfth of core discharged for refueling

	Recycle of All Fuel						Bred Fuel Recycled Only Once, Feed Not Recycled		Bred Fuel All Recycled, Feed Not Recycled	
	Case 8	Case 11	Case 9	Case 1	Case A-8	Case A-9	Case 2	Case A-2	Case 3	Case A-3
Equilibrium or last-cycle data										
Exposure, Mwd/MT of U + Th	45,000	52,500	58,000	75,600	43,700	57,200	58,500	60,000	68,600	67,800
Fuel lifetime, full-power days	1860	2170	2380	3230	1780	2330	2380	2410	2830	2780
Fabrication and processing plant size for 15,000 Mw(e), MT of U + Th per year	222	190	172	133	228	174	171	167	146	147
Fabrication cost with new BeO (first core only), $/kg of U + Th	170	172	177	184	169	177	177	178	182	182
Fabrication cost, using recycle BeO, $/kg of U + Th	106	110	112	120	105	112	112	113	118	118
Processing cost, $/kg of U + Th	86	97	106	130	83	104	114	116	130	127
Fuel shipping costs (fabricated + irradiated), $/kg of U + Th	25.65	25.65	25.65	25.65	25.65	25.65	25.65	25.65	25.65	25.65
Preexposure fuel holdup, days	182	182	182	182	182	182	182	182	182	182
Postexposure holdup through processing, days	211	211	211	211	211	211	211	211	211	211
Initial fissile inventory, kg	3570	3570	3570	3570	3570	3570	3560	3560	3560	3560
Fuel inventory cost of first core, dollars	44.0×10^6	44.0×10^6	44.0×10^6	44.0×10^6	44.0×10^6	44.0×10^6	43.9×10^6	43.9×10^6	43.9×10^6	43.9×10^6
Fabrication cost of first core, dollars	17.0×10^6	17.2×10^6	17.7×10^6	18.5×10^6	16.9×10^6	17.7×10^6	17.7×10^6	17.8×10^6	18.3×10^6	18.3×10^6
Fuel cycle cost, mills/kwhr(e)										
Fabrication	0.309	0.266	0.246	0.225	0.284	0.239	0.234	0.226	0.229	0.225
Burnup + losses[c]	0.234	0.262	0.294	0.341	0.201	0.260	0.255	0.235	0.273	0.242
Processing	0.200	0.196	0.191	0.179	0.189	0.179	0.191	0.185	0.201	0.196
Shipping	0.062	0.053	0.047	0.039	0.059	0.045	0.044	0.042	0.040	0.040
U + Th inventory	0.636	0.641	0.645	0.650	0.621	0.622	0.630	0.638	0.628	0.609
Interest on fabrication	0.204	0.191	0.188	0.214	0.182	0.190	0.190	0.189	0.207	0.200
Interest on processing	-0.020	-0.024	-0.025	-0.022	-0.025	-0.026	-0.028	-0.030	-0.024	-0.029
Interest on shipping	0.001	0	0	0.002	-0.001	0	0	0	0.001	0
Total[d]	1.626	1.585	1.586	1.628	1.510	1.509	1.516	1.485	1.555	1.483
Fuel discard penalty,[e] discounted							0.117	0.097	0.090	0.073
Total							1.633	1.582	1.645	1.556
Fuel sale at reduced value,[f] discounted							-0.068	-0.056	-0.053	-0.043
Net fuel cycle cost	1.63	1.58	1.59	1.63	1.51	1.51	1.56	1.53	1.59	1.51

[a]Cases with numbers preceded by A are for higher epithermal eta values for U^{233}; others are for the lower value.

[b]Cases with numbers preceded by B are for a Be-to-Th ratio of 32; those preceded by C are for no BeO in core; all others are for a Be-to-Th ratio of 28.

[c]Losses = 1% in processing +0.2% in fabrication.

[d]Full $12/g of fissile material credit allowed for highly irradiated discharged fuel.

[e]No value allowed for highly irradiated discharged fuel (but final core loading credited at full value).

[f]Sale of highly irradiated discharged fuel at $7/g of fissile material.

Table 12.7. (continued)

	Initial Loading and All Bred Fuel Only Recycled Once								
	Case 4	Case 10	Case 5	Case B-10	Case A-4	Case A-10	Case A-5	Case AB-5	Case AC-10
Equilibrium or last-cycle data									
Exposure, Mwd/MT of U + Th	30,100	48,500	58,500	29,000	30,500	47,600	58,900	52,300	37,400
Fuel lifetime, full-power days	1210	1860	2380	1250	1236	1836	2388	1812	864
Fabrication and processing plant size for 15,000-Mw(e), MT of U + Th per year	332	206	171	345	323	207	167	189	264
Fabrication cost with new BeO (first core only), $/kg of U + Th	160	171	177	172	160	171	178	187	86[g]
Fabrication cost, using recycle BeO, $/kg of U + Th	97	107	112	102	98	107	113	115	86[g]
Processing cost, $/kg of U + Th	66	97	106	64	67.50	98	107	105	79.60
Fuel shipping costs (fabricated + irradiated), $/kg of U + Th	25.65	25.65	25.65	25.65	25.65	25.65	25.65	25.65	25.65
Preexposure fuel holdup, days	182	182	182	182	182	182	182	182	182
Postexposure holdup through processing, days	211	211	211	211	211	211	211	211	211
Initial fissile inventory, kg	3380	3466	3570	2810	3380	3466	3570	2910	2600
Fuel inventory cost of first core, dollars	41.8×10^6	42.8×10^6	44.0×10^6	34.8×10^6	41.8×10^6	42.8×10^6	44.0×10^6	36.0×10^6	31.9×10^6
Fabrication cost of first core, dollars	16.0×10^6	17.1×10^6	17.7×10^6	15.0×10^6	16.0×10^6	17.1×10^6	17.8×10^6	16.4×10^6	4.6×10^6
Fuel cycle cost, mills/kwhr(e)									
Fabrication	0.331	0.262	0.234	0.305	0.323	0.257	0.233	0.256	0.208
Burnup + losses[c]	0.172	0.223	0.252	0.177	0.158	0.197	0.227	0.194	0.304
Processing	0.195	0.197	0.191	0.164	0.194	0.194	0.179	0.194	0.184
Shipping	0.077	0.053	0.044	0.067	0.074	0.052	0.044	0.048	0.060
U + Th inventory	0.612	0.628	0.651	0.522	0.591	0.597	0.606	0.508	0.469
Interest on fabrication	0.160	0.184	0.192	0.157	0.155	0.174	0.187	0.168	0.056
Interest on processing	-0.024	-0.025	-0.025	-0.017	-0.030	-0.034	-0.032	-0.030	-0.022
Interest on shipping	-0.004	0	0	-0.001	-0.005	-0.002	-0.001	-0.002	-0.003
Total[d]	1.519	1.522	1.539	1.374	1.460	1.435	1.443	1.336	1.256
Fuel discard penalty,[e] discounted	0.163	0.074	0.051	0.118	0.082	0.025	0.020	0.026	0.108
Total	1.682	1.596	1.590	1.492	1.542	1.460	1.463	1.362	1.364
Fuel sale at reduced value,[f] discounted	-0.095	-0.043	-0.030	-0.069	-0.049	-0.014	-0.012	-0.015	-0.063
Net fuel cycle cost	1.59	1.55	1.56	1.42	1.49	1.45	1.45	1.35	1.30

[g]No BeO in core.

Since fuel-cycle costs have generally been computed in the past on the basis of a single "equilibrium" cycle, a comparison is made in Table 12.8 of the results of such a cost computation with the results for a 30-year history. In this case the "equilibrium" costs were calculated from the final cycle in the 30-year history, which is not far from true equilibrium in terms of isotopic concentrations. The differences between the 30-year costs and the equilibrium costs are identifiable with the less favorable conditions existing when the reactor is started up with U^{235} feed rather than U^{233} and with the necessity for doing more fuel fabrication and processing at the beginning. The effect of the first few cycles is heavily weighted by the high "present value" of the early years of operation. This shows up most strongly in the fabrication interest because the high initial cost of the BeO (it is later recycled at a lower cost) is included in the fabrication charge.

Table 12.8. An Example of the Comparison of 30-Year Fuel-Cycle Costs with Equilibrium Fuel-Cycle Costs in HTGR

	30-Year Cost (mills/kwhr)	Equilibrium Cost (mills/kwhr)
Fabrication	0.26	0.21
Burnup and losses	0.25	0.24
Processing	0.20	0.19
Shipping	0.05	0.04
Inventory	0.63	0.49
Working capital	0.16	0.04
Total	1.55	1.21

A calculation made for a simple case illustrates the manner in which consideration of 30-year life alters fabrication and processing costs and the associated interest costs. The fuel was assumed to have a six-year life; fuel elements are purchased one-half year before use and processed one year after discharge. A full core was loaded at startup, with refueling (one-twelfth of the core) begun two years later and continued semi-annually. Credit for remaining fuel life and a charge for processing were made at the end of the 30 years in the manner described earlier.

Considering one-twelfth of the core to be a fuel unit and taking fabrication and processing costs of unity for each fuel unit, the following results are obtained:

	Fabrication	Processing
Treatment of 30-year reactor life		
Number of units charged against 60 units of power	62.5	62.5
Direct cost per unit of power	1.04	1.04
Interest per unit of power		
At 10% per year	0.48	
At 6% per year		−0.16
Interest plus direct cost per unit of power	1.52	0.88
Treatment as 6-year-life batch without compounding interest		
Number of units charged per unit of power	1.0	1.0
Interest per unit of power		
At 10% per year	0.35	
At 6% per year		−0.24
Interest plus direct cost per unit of power	1.35	0.76

For the example used, the fabrication interest is 38% more in the 30 years than in the batch treatment, and the processing interest, a credit, is 33% less; the total costs for fabrication and processing in the 30-year case are 13 and 16% more. These percentages would be different for other cycle times or fuel-management schemes; in particular, the fabrication interest would be considerably amplified if the cost of purchasing the BeO for the initial fuel elements was considered.

The values plotted in Fig. 12.5 are for the reference design condition of a beryllium-to-thorium ratio of 28, which appears to be less than optimum. Case AB-5, computed with a ratio of 32, has a fuel-cycle cost that falls below the curve in Fig. 12.5, and case AC-10, which is for an all-graphite core, falls even further below. In addition, General Atomic has provided equilibrium physics calculations, which, when combined with the economic factors from this study, give the lower costs shown in Table 12.9 for higher beryllium-to-thorium ratios or no BeO. Lower fuel-cycle costs for the all-graphite core result largely from eliminating the cost of the BeO.

Thus it appears that lower fuel-cycle costs can be obtained in an HTGR by the use of less BeO than specified in the reference design (or none at all), although the breeding ratio and hence the fuel utilization is reduced. At higher beryllium-to-thorium ratios, or with no BeO, the power peaking becomes more unfavorable. This may be offset by a difference in the element design, but thermal analyses to show the effect quantitatively are not available.

Table 12.9. Equilibrium Fuel-Cycle Cost for Various Fuel Elements in HTGR (6-Year Fuel Life)[a]

	Fuel-Cycle Cost (mills/kwhr)
BeO spines	
Be/Th = 28	1.37
Be/Th = 32	1.26
Be/Th = 36	1.18
Graphite spines	
C/Th = 140	1.19
C/Th = 175	1.09

[a]Based on General Atomic's physics calculations.

The highly irradiated fuel discharged from the HTGR at the end of each cycle is a mixture of U^{233}, U^{234}, U^{235}, and U^{236} and, in addition, contains U^{232}. Because of the U^{236} and U^{232}, its value should be less than that of fully enriched U^{235}, but its actual worth is quite uncertain. A value of \$7/g of fissile material was arbitrarily allowed in this study. If the fuel were discarded without sale, the penalty would be less than 0.07 mill/kwhr before deducting an offsetting saving of about 10% of the processing cost.

Taking into consideration the uncertainty in the epithermal cross sections of U^{233} and the variety of possible fuel-cycle conditions, the 30-year average fuel-cycle cost of the HTGR appears to be between 1.3 and 1.4 mills/kwhr.

Fuel-Cycle Cost for SGR

Fuel-cycle costs as a function of enrichment were calculated for the SGR based on the physics results presented in Chapter 11. The complete breakdown of costs of three 30-year history cases are presented in Table 12.10. Equilibrium cycle calculations indicated that there is approximately 0.15 mill/kwhr cost penalty associated with the nonequilibrium initial cycles. This penalty could possibly be reduced through optimizations on the startup cycles.

As noted in Chapter 11, there is a discrepancy between physics calculations made by ORNL and AI that is due in part to differences in reactor design. Fuel costs calculated for a representative case based on

Table 12.10. Thirty-Year Fuel-Cycle Costs of a 1019-Mw(e) Sodium-Graphite Reactor (SGR)

Fixed charge on capital investment in fabrication and processing plants: 22% per year
Fabricating days per year: 260
Processing days per year: 155
Plutonium value: $7 per gram of fissile material
One-sixth of core discharged for refueling

Case	1A-1	1A-2	1A-3
Equilibrium or last-cycle data			
Exposure, Mwd/MT of U	16,400	19,000	22,000
Fuel lifetime, full-power days	488	570	661
Feed enrichment, wt % U^{235}	2.87	3.14	3.45
Fabrication and processing plant size for 15,000 Mw(e), MT of U per year	611	524	451
Fabrication cost, $/kg of U	33	37.50	40.50
Processing cost, $/kg of U	33	38.00	43.30
Fuel shipping costs (fabricated + irradiated), $/kg of U	4.28	4.35	4.42
Preexposure fuel holdup, days	95	96	98
Postexposure holdup through processing, days	161	162	163
Initial fissile inventory, kg	1674	1701	1728
Fuel inventory cost of first core, dollars	13.2×10^6	13.5×10^6	13.8×10^6
Fabrication cost of first core, dollars	2.30×10^6	2.62×10^6	2.83×10^6
Fuel-cycle cost, mills/kwhr(e)			
Fabrication	0.200	0.196	0.184
Burnup + losses[a]	0.962	0.968	0.977
Processing	0.197	0.195	0.192
Shipping	0.026	0.022	0.020
Uranium inventory	0.244	0.253	0.264
Interest on fabrication	0.030	0.034	0.036
Interest on processing	−0.012	−0.012	−0.013
Interest on shipping	0.001	0.001	0.002
Pu credit	−0.151	−0.141	−0.133
Pu inventory	0.016	0.017	0.017
Net fuel-cycle cost	1.51	1.53	1.55

[a]Losses = 1% in processing + 0.2% in fabrication.

AI physics indicated about 0.1 mill/kwhr lower fuel costs resulting from differences in the physics.

Allowing for differences in ORNL and AI physics calculations, the best estimate for the 30-year fuel-cycle cost of the sodium-graphite reactor is in the range 1.4 to 1.5 mills/kwhr.

Summary

A near-optimum example of the fuel-cycle cost breakdown for each concept based on ORNL physics calculations (for a 30-year life) is presented in Table 12.11. Included in the table are ranges in fuel-cycle costs within which the minimum cost for each concept is believed to lie. Use of a range rather than a single value acknowledges uncertainties in cross sections and unresolved differences in the physics calculations performed by ORNL and by the sponsors of the concepts. The specific reasons for the numbers cited are discussed in the above sections that consider the various reactors. No allowances are included in the ranges cited for the uncertainties in fabrication and processing costs. While the procedure

Table 12.11. Summary Comparison of Fuel-Cycle Cost[a]

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Case	7	B-3	E-1	G-4	AB-5	1A-2
Last-cycle exposure, Mwd/MT	21,050	22,600	11,100	28,500	52,300	19,000
Unit fabrication cost, $/kg	53	60	21.9	37.8	115[b]	37.5
Unit processing cost, $/kg	31.6	37	16.5	40.2	105	38
Fuel-cycle cost breakdown, mills/kwhr(e)						
Fabrication	0.34	0.28	0.31	0.22	0.26	0.20
Burnup and losses	0.99	0.70	0.81	0.41	0.20	0.97
Processing	0.20	0.17	0.23	0.24	0.19	0.19
Shipping	0.03	0.03	0.03	0.04	0.05	0.02
Inventory	0.24	0.63	0.10	0.39	0.51	0.27
Interest on fabrication	0.08	0.08	0.04	0.05	0.17	0.03
Interest on processing	−0.01	−0.01	−0.01	−0.01	−0.03	−0.01
Plutonium credit	−0.20		−0.32			−0.14
Total	1.67	1.88	1.19	1.34	1.35	1.53
Estimated range	1.7–1.8	1.6–1.9	1.0–1.2	1.2–1.4	1.3–1.4	1.4–1.5

[a]Based on 22% per year fixed charges on fabrication and processing plants, 10% per year fixed charges on fuel and fuel element fabrication working capital, $7/g of fissile Pu and highly irradiated discharged uranium, $8/lb of U_3O_8; cost breakdowns based on ORNL 30-year-life physics calculations.

[b]For recycle of BeO.

used in estimating unit costs was designed to make the relative values as accurate as possible, this is clearly an imperfect process, and the uncertainty should be recognized in comparing costs for the various concepts.

The values in Table 12.11 reflect the effect of the large-size fabrication and processing plants assumed for this study. Costs associated with fabrication and processing range from 0.41 mill/kwhr for the SGR to 0.61 for the PWR. As percentages of the net fuel-cycle costs, these costs vary from 28% for the SSCR to 48% for the HWR-U. Although the HTGR has relatively high unit costs, they are offset by high burnup and high thermal efficiency.

An effect of the lowering of importance of fabrication and processing costs is an increase in relative importance of the costs associated with owning and consuming uranium, particularly for the 10% inventory charge used for the reference conditions. The HWR-U achieves the lowest fuel-cycle cost (if D_2O costs are excluded) by virtue of using low-enrichment uranium with its low cost per gram of U^{235}; and the HTGR and HWR-Th offset relatively high inventory costs with low burnup cost resulting from good neutron economy and, in the case of the HTGR, high thermal efficiency. It is worth noting here that the burnup cost of the HWR-U may be higher than would be expected because neutron absorptions in the heavy-wall pressure tube offset somewhat the savings from using heavy water as the moderator.

Although the HWR-Th gains 0.08 mill/kwhr in net burnup cost over the HWR-U, it suffers a penalty of 0.31 mill/kwhr in inventory cost because of the use of fully enriched uranium and because of a lower power density. The PWR and SGR have high burnup costs because of poorer neutron economy than the other systems, and the effect on the SGR would be worse were it not for the benefit of its high thermal efficiency. While the SSCR achieves somewhat lower burnup costs than the PWR and SGR, the saving is more than offset by the high inventory cost that results from using fully enriched uranium.

Fuel inventory costs range from 0.10 mill/kwhr for the HWR-U to 0.63 for the SSCR. Because of the relatively low fabrication and reprocessing costs and the 10% inventory charge on fuel, the fuel cycles generally tend to optimize at lower burnups than those cited for the reference designs by the concept sponsors.

The statement of ground rules in Chapter 2 recognizes that several of the cost factors are quite uncertain. Differences from the reference values that affect the costs of one reactor relative to another are particularly important in comparing concepts. Table 12.12 shows the effect of varying the interest charge and the plutonium value through the range stated in the ground rules. The effect of changing the price of U_3O_8 over a range of from \$4 to \$16/lb is also shown, with the values of U^{233} and plutonium varied in proportion to the value of fully enriched U^{235}. In considering the effects of these perturbations, it must be recognized that the fuel cycles have not been reoptimized for the changed conditions. The penalties associated with unfavorable changes would be reduced by reoptimization and the gains from favorable changes increased.

Altering the interest rate has the greatest effect on the SSCR and HTGR. The HWR-U is least affected because of its very low inventory investment cost.

Table 12.12. Effect of Variation in Economic Factors on 30-Year Fuel-Cycle Cost[a] Based on ORNL Physics Results

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Fuel-cycle cost for reference conditions,[b] mills/kwhr	1.67	1.88	1.19	1.34	1.35	1.53
Effect of change in fuel inventory and fabrication interest charges, mills/kwhr						
At 6% per year	1.54	1.60	1.14	1.16	1.08	1.41
At 12% per year	1.74	2.02	1.22	1.42	1.49	1.58
Effect of change in uranium ore cost (with Pu and U^{233} values changed in proportion to fully enriched uranium), mills/kwhr						
At \$4/lb of U_3O_8 and \$9.54/g of U^{235}	1.37	1.60	0.94	1.17	1.20	1.24
At \$16/lb of U_3O_8 and \$16.23/g of U^{235}	2.14	2.34	1.67	1.62	1.60	2.00
Effect of change in value of plutonium and highly irradiated uranium, mills/kwhr						
At \$3/g of fissile material	1.77	1.88	1.29	1.34	1.36	1.60
At \$12/g of fissile material	1.55	1.88	0.97	1.34	1.34	1.44

[a]Values given are perturbations of the base cases without reoptimization.

[b]Based on 22% per year fixed charges on fabrication and processing plants, 10% per year fixed charges on fuel and fuel element fabrication working capital, \$7/g of fissile material for Pu and highly irradiated discharged uranium, \$8/lb of U_3O_8.

A doubling of the uranium ore cost would increase the fuel-cycle cost of the HTGR by 0.25 and the HWR-Th by 0.29 mill/kwhr. The cost for each of the other concepts would increase by about 0.47 mill/kwhr.

Variation in the value of plutonium affects only the uranium-cycle reactors. Increasing the value to \$12/g would decrease the fuel-cycle cost for the PWR and SGR about 0.1 mill/kwhr and that for the HWR-U about 0.2. Reducing the value to \$3 would increase the cost of all three about 0.1 mill/kwhr. The HWR-U change is limited as the plutonium value falls to that associated with storing the discharged fuel instead of reprocessing it.

Reducing the value of the highly irradiated uranium discharged from each HTGR cycle from the \$7/g of fissile material reference value down to \$3/g only increases the fuel-cycle cost for the case illustrated by 0.01 mill/kwhr, and increasing it to \$12/g reduces the cost by about a like amount. For other cases and fuel-management schemes the change would be higher, but the penalty incurred by a change to \$3/g would not exceed 0.05 mill/kwhr in the most extreme case calculated.

13. POWER COSTS

Power costs were computed using capital costs obtained from Chapter 6, operating costs from Chapter 7, and fuel-cycle costs from Chapter 12. All costs are based on an 80% plant factor and a fixed charge rate on the reactor plant of 12% per year. The results for the reference conditions are summarized in Table 13.1.

The fuel-cycle costs presented apply to a 30-year reactor life, with costs averaged using the present-worth procedure. Uncertainties in the physics, as discussed in Chapters 11 and 12, are acknowledged by showing a possible range of costs rather than a single value.

A single number representing the best estimate is presented for the capital costs, although as shown graphically in Fig. 6.1, there is considerable uncertainty in the estimates. This uncertainty does not refer to differences from the "price" of a plant but to differences in the cost of one concept relative to that of another. Considering the items specifically identified in Chapter 6, it appears that the capital cost of any plant relative to the others could be as much as 0.1 mill/kwhr lower than the value given in Table 13.1. However, only the heavy-water reactors and the HTGR are likely to be much higher than the estimate, with the HWR-U and HWR-Th possibly going up 0.1 mill/kwhr, while the HTGR could increase by 0.2 mill/kwhr. Although the same operating cost is used for all plants, there will clearly be some variation, but insufficient information is available at present to make a distinction.

In spite of the acknowledged uncertainties associated with the cost values, the totals presented in Table 13.1 represent the best estimates obtained for each concept in this study. A distribution of costs for each concept into fixed and variable contributions is given in Table 13.2.

Table 13.1. Power Costs for Reference Economic Conditions[a]

	Power Costs (mills/kwhr)					
	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Capital cost	2.3	2.1	2.1	2.3	2.0	2.2
Operating cost	0.3	0.3	0.3	0.3	0.3	0.3
Fuel-cycle cost	1.7–1.8	1.6–1.9	1.0–1.2	1.2–1.4	1.3–1.4	1.4–1.5
Heavy-water cost		0.1	0.5	0.6		
Total power cost	4.3–4.4	4.1–4.4	3.9–4.1	4.4–4.6	3.6–3.7	3.9–4.0

[a]Based on 12% per year fixed charges on reactor plant, 22% per year fixed charges on fabrication and processing plants, 10% per year fixed charges on fuel and fuel element fabrication working capital, \$7/g of fissile material for Pu and highly irradiated discharged uranium, \$8/lb of U_3O_8. Heavy-water cost includes 10% per year for inventory plus 2% per year for makeup.

Table 13.2. Power Cost Breakdown into Fixed and Incremental Contributions Based on ORNL Physics Giving Near-Optimum Power Cost

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
With Reference Ground Rules (Private Utility)[a]						
Annual fixed charges, $/kw	20	23	21	25	21	20
Incremental cost, mills/kwhr	1.4	1.2	1.1	0.9	0.7	1.2
Total power cost at 0.8 load factor, mills/kwhr	4.3	4.4	4.1	4.5	3.7	4.0
With Public Utility Financing[b]						
Annual fixed charges, $/kw	13	14	13	15	13	12
Incremental cost, mills/kwhr	1.4	1.2	1.1	0.9	0.7	1.2
Total power cost at 0.8 load factor, mills/kwhr	3.3	3.2	3.0	3.0	2.5	2.9

[a] 12% per year fixed charges on reactor plant; 10% per year fuel and fuel element fabrication interest.

[b] 7% per year fixed charges on reactor plant; 5% per year fuel element fabrication interest. Not reoptimized for changed conditions.

This type of breakdown is particularly valuable in considering use of a plant in a specific power system. Also shown in Table 13.2 are estimated costs for a public utility. The values presented for public utility financing represent only perturbations of the reference cases and have not been reoptimized for the changed conditions.

14. FUEL UTILIZATION

One of the primary motivations for developing reactors with high conversion ratios is that such reactors make more efficient use of mined uranium than systems with low neutron economy. The attainment of an equilibrium conversion ratio of 0.95, for example, results in a sixfold increase in the energy yield from a given quantity of natural uranium compared with that for a conversion ratio of 0.7, which is typical of slightly enriched light-water reactors. An analysis of the fuel utilization performance of advanced converters therefore provides an additional basis for evaluating the merits of each type of reactor. In carrying out such an evaluation, however, the point should be made that the specific designs considered in this study were aimed at minimizing power costs under the given set of economic ground rules. The conversion ratios and corresponding fuel utilization performances of these designs are therefore not representative of the best that can be achieved in any given case. It should also be emphasized that the results are not comparable on a relative basis because for some concepts designers paid more attention to neutron economy than for others. Thus the results summarized below provide only a qualitative indication of fuel utilization for concepts designed to operate most economically under the specified conditions.

The fissile inventory and consumption requirements used as a basis for calculating fuel utilization performances are given in Table 14.1 (taken from Table 11.28). Data for a hypothetical "advanced" converter to be used as a basis for comparison are also shown in Table 14.1.

The future installed capacity of the U.S. nuclear industry, as taken from the November 1962 "AEC Report to the President" and extrapolated beyond the year 2000, is shown in Table 14.2. Data in Table 14.1 were then combined with these projected capacities to determine the total amount of natural uranium that must be mined or committed for the 30-year life of each new plant. The results are summarized in Table 14.3. In each

Table 14.1. Fuel Requirements of Converter Reactors

Concept	Case	Fuel Requirement [kg/Mw(e)]			
		Initial Fissile Inventory	Annual Net Fissile Makeup	Thirty-Year Net Fissile Makeup	Final Fissile Loading
PWR	7	2.1	0.63	18.8	1.4
SSCR	B-3	3.3	0.39	11.8	2.7
HWR-U	E-1	0.6	0.79	23.6	0.49
HWR-Th	G-4	1.4	0.25	7.7	1.67
HTGR	AB-5	2.9	0.11	3.4	1.97
SGR	1A-2	1.7	0.58	17.4	1.60
"Advanced" converter		1.0	0.05	1.5	0.95

Table 14.2. Postulated Future Installed Capacity of U.S. Nuclear Industry

Year	Industry Capacity (millions of kw)			
	Installed	New	New Plus Replacement	Cumulative Installed Total
1970	5	5	5	5
1980	40	35	35	40
1990	210	170	170	210
2000	730	520	525	735
2010	1500	770	805	1540
2020	2300	800	970	2510
2030	3200	900	1420	3935
2040	4000	800	1570	5540

Table 14.3. Cumulative Natural Uranium Mined and Committed

Reactor	Millions of Short Tons of U_3O_8				
	1980	1990	2000	2010	2020
PWR	0.21	1.09	3.78	7.75	12.42
SSCR	0.18	0.91	3.10	6.13	9.57
HWR-U	0.22	1.13	3.93	8.17	13.22
HWR-Th	0.11	0.58	1.97	3.93	6.16
HTGR	0.06	0.32	1.06	2.02	3.06
SGR	0.20	1.06	3.67	7.50	12.01
"Advanced converter"	0.03	0.14	0.48	0.91	1.38

case it was assumed that the industry consists entirely of the reactor concept in question. A plot of these data in Fig. 14.1 provides an indication of the potential exhaustion dates of any given amount of natural uranium ore for an industry based on each type of reactor. Similarly, exhaustion dates for combinations of reactors can be found from these curves by multiplying the requirement for each reactor type by its fraction of total capacity. An industry consisting of PWR's, for example, would exhaust or commit 800,000 short tons of U_3O_8 by the year 1988. An HWR-Th or HTGR reactor system, however, would extend this date to 1996, while the hypothetical "advanced" converter would not utilize or commit 800,000 tons of ore until 2005. An industry consisting of 50% PWR's and 50% advanced converters, on the other hand, would run out of this amount

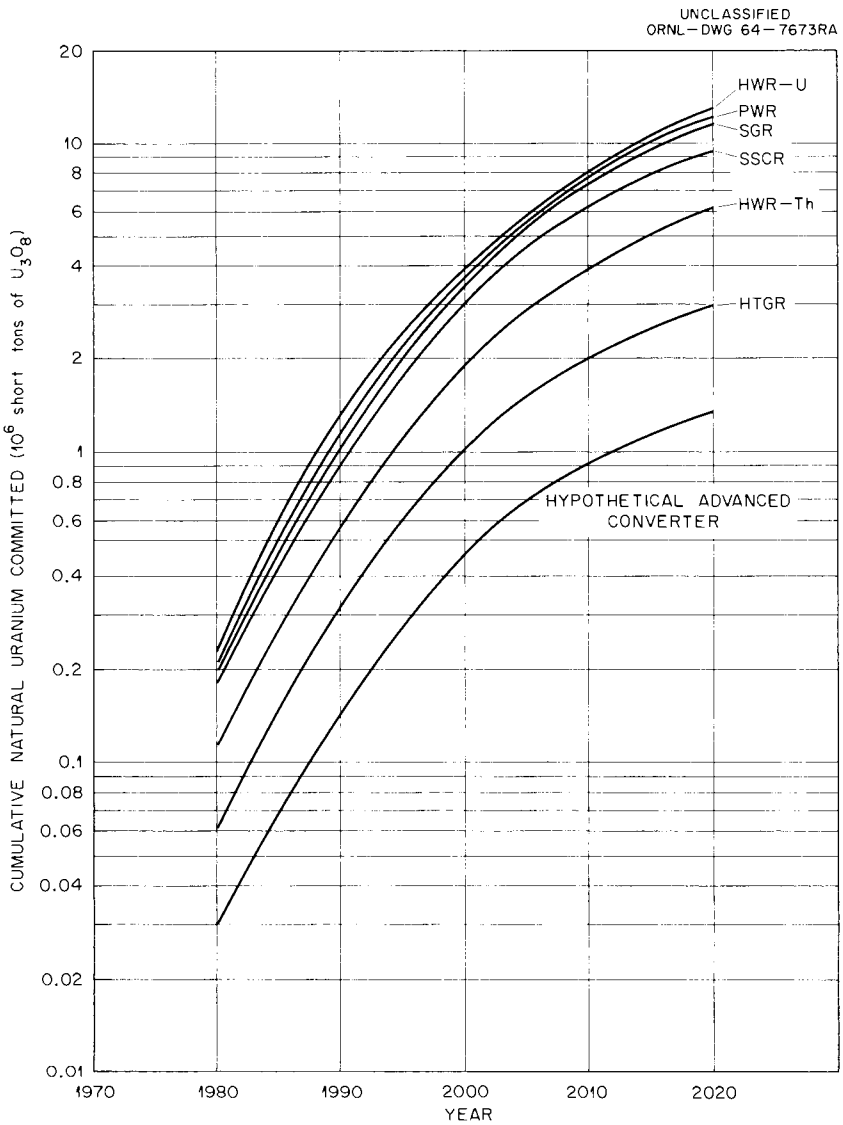

Fig. 14.1. Natural Uranium Committed as a Function of Time.

of natural uranium by the year 1992. This indicates that in order to appreciably extend the date at which the low cost natural uranium might be exhausted, most of the future nuclear capacity would have to consist of high-neutron-economy converters.

15. SUMMARY AND CONCLUSIONS

This report presents the results of an evaluation in which the power costs of five advanced converter concepts and a 1000-Mw(e) pressurized-water reactor were estimated and compared. In addition, the engineering performance requirements of the fuel elements and the remainder of the reactor plant were reviewed and the status of the technology established.

To provide a basis for the cost estimates, the time period and economic context in which the reactors were to be evaluated was established. Briefly, the plants were assumed to begin operating in the 1970's and, using present-worth accounting, their histories were followed over a 30-year life. Of particular importance in the power cost estimates are the assumptions that (1) the low-cost uranium reserves are not exhausted during the life of the reactors; (2) the fabrication and processing plants are of a size that will support 15,000-Mw(e) installed capacity of a given type of reactor; (3) there is private ownership of fuel, reactor plant, and fuel fabrication and reprocessing facilities; and (4) the thorium reactors are started up with U^{235} and the irradiated fuel is recycled, but plutonium is sold without recycle.

The methods followed in the study are described in the preceding parts of this report and the results of each aspect of the evaluation are summarized at the ends of the appropriate chapters. The major results are further summarized below.

Power Cost and Fuel Utilization

The plant capital costs, the fuel-cycle costs, the power costs, and the fuel utilization estimated for the various concepts are given in Table 15.1. A range is shown for the fuel-cycle costs (and reflected in the power costs) to acknowledge uncertainties in the physics calculations and differences between ORNL calculations and those of the sponsors of the concepts.

The low capital costs of the HTGR are attained largely through use of a prestressed-concrete reactor vessel that contains the entire primary system, and lack of experience with such vessels and equipment for the "all-inside" design makes the HTGR estimate the most uncertain. Higher costs for the SGR result from the additional cost of equipment required for the intermediate sodium coolant system. The SSCR cost is lower than that of the PWR because of the benefits attained from a high core power density. Higher costs for the HWR-Th over the HWR-U result from a lower core power density and the cost of an on-stream refueling machine.

Because of the large-size fuel fabrication and processing plants assumed for this study, costs associated with fabrication and processing range from only 28 to 48% of the fuel-cycle costs for the reference cases. An effect of lowering the importance of these costs is to increase the importance of fuel burnup and inventory costs, particularly for a 10% inventory charge. Consequently the fuel cycles generally tend to optimize at lower burnups than those cited for the reference designs by the concept sponsors.

Table 15.1. Summary of Results of Evaluation of Advanced Converters

	Pressurized-Water Reactor (PWR)	Spectral-Shift-Control Reactor (SSCR)	Heavy-Water Reactor, Uranium Fueled (HWR-U)	Heavy-Water Reactor, Thorium Loaded (HWR-Th)	High-Temperature Gas-Cooled Reactor (HTGR)	Sodium-Graphite Reactor (SGR)
Fuel cycle	U	Th	U	Th	Th	U
Initial fissile inventory, kg/Mw(e)	2.1	3.3	0.6	1.4	2.9	1.7
30-year net fissile makeup, kg/Mw(e)	19	12	24	7.7	3.4	17
Plant capital cost, $/kw(e)	133	125	125	134	117	130
Heavy-water inventory, $/kw(e)		6	27	33		
Fuel-cycle cost, mills/kwhr	1.7–1.8	1.6–1.9	1.0–1.2	1.2–1.4	1.3–1.4	1.4–1.5
Power cost, mills/kwhr	4.3–4.4	4.1–4.4	3.9–4.1	4.4–4.6	3.6–3.7	3.9–4.0

The HWR-U achieves the lowest fuel-cycle cost (if D_2O costs are excluded) through use of low-enrichment uranium with its low cost per gram of U^{235}; and the HWR-Th and HTGR, which have the next lowest fuel cycle costs, benefit from low burnup costs resulting from good neutron economy. The HTGR receives additional benefit from high thermal efficiency. Both the PWR and SGR suffer from high burnup costs, but the SGR receives some offsetting advantage from its high thermal efficiency. The SSCR achieves somewhat lower burnup costs than the PWR and SGR, but the saving is more than balanced by the high inventory cost that results from using fully enriched uranium.

The reference economic conditions used in the cost estimates included a 10% annual charge on fuel and fuel fabrication investment, an \$8/lb of U_3O_8 ore cost, and a fissile plutonium value of \$7/g. Use of a lower interest rate would be of greatest benefit to the SSCR and HTGR. Doubling the ore cost would increase the fuel-cycle costs of the HTGR and HWR-Th about 0.2 to 0.3 mill/kwhr and the costs of the other reactors by nearly 0.5 mill/kwhr. Increasing the plutonium value to \$12/g would decrease the fuel-cycle cost of the PWR and SGR by 0.1 mill/kwhr and that of the HWR-U by 0.2 mill/kwhr; reducing the value to \$3/g would increase the cost of each of the three by 0.1 mill/kwhr. It should be noted that the specific concepts analyzed in this study were generally optimized for minimum power costs, and the fuel utilization values shown in Table 15.1 are not representative of the best that could be achieved with the concept.

While the objective of this study was to compare the various reactor concepts on the basis of their economic performance, the procedure actually followed was to evaluate specific designs proposed by sponsors of the concepts. In some cases the reference design may not represent the lowest cost achievable with a concept. Particular examples are the heavy-water reactors in which use of coolants other than pressurized heavy water would probably result in lower power costs.

Engineering Performance of the Core

Evaluation of the thermal designs of the various reactors indicated that all appear to be adequate, with the reservation that the power peaking factors were not checked by physics calculations and the enthalpy rise factor claimed for the PWR has not yet been demonstrated. With regard to fuel element requirements, the PWR and SSCR appear to be designed on the basis of current technology.

All the remaining reactors are designed on the basis of extrapolations of fuel element technology well beyond existing data. Tests of the nested fuel elements for the heavy-water reactors have not progressed to the burnups required at the specified heat ratings, and neither the largest size tubes nor assemblies of tubes have yet been tested. Research and testing will be required to demonstrate successful performance of the vented fuel concept proposed for the SGR, the ability to maintain low enough impurity levels in the sodium coolant to prevent corrosion of the Zircaloy cladding, and satisfactory performance of the UC fuel at the temperature and burnup proposed. The coated particles in the HTGR fuel

are required to perform satisfactorily in larger sizes and to higher burnups than have been tested, and operation of graphite fuel elements for a six-year life of course remains to be demonstrated.

Engineering Performance of the Reactor Plant

Both the PWR and the SSCR require large heavy-walled pressure vessels that are beyond present experience, but the problems associated with their construction appear to be solvable. The heavy-water reactors use Zr-2.5% Nb alloy pressure tubes on which corrosion and creep information are favorable but which require excessive extrapolation to predict a 30-year behavior. An on-line refueling machine specified for the HWR-Th will require high reliability under severe conditions. Adequately low losses of heavy water from the SSCR and the heavy-water reactors can probably be achieved, but care with closures, seals, and leakage recovery systems will impose some burdens during operation and maintenance.

The SGR design represents a departure from previous plants in the use of a calandria concept with 0.025-in.-thick stainless steel process tubes. This concept appears to bear an element of risk, since a break in one of the tubes could require a prolonged shutdown for repair.

Feasibility of the HTGR concept proposed depends largely on successful demonstration of the prestressed-concrete reactor vessel that houses the entire primary system, including the core, helium-circulating blowers, and steam generators. While French and British experience with concrete reactor vessels is applicable, and both the French and British have "all-inside" plants under construction, successful performance of vessels of the HTGR type remains to be demonstrated.

A preliminary analysis of plant availability indicates that a high degree of reliability will be required of all reactor equipment, particularly for the HWR-U, HTGR, and SGR, which have more frequent refueling shutdowns. The reference designs for each of these concepts are marginal in meeting a 0.90 availability factor, even if there is no unscheduled downtime for reactor equipment maintenance.

The state of development of the systems and components required for the operation of the reactors is far from equal. The light-water-cooled reactors have the most extensive operating experience to call on. Heavy-water reactors have the advantage of NPD, CVTR, PRTR, and HWCTR experience, with the commissioning of the 200-Mw(e) CANDU reactor anticipated soon. The SGR has, as a base, both the SRE and HNPF and additional background in sodium technology from Fermi, EBR-1, and EBR-2. Until operation of the Peach Bottom, Dragon, EGCR, and Oldbury stations is under way, the technology of the HTGR must be considered the least firmly established.

APPENDICES

Appendix A

BASIS FOR FIXED CHARGES ON NUCLEAR PLANTS

The annual fixed charge applicable to depreciating capital investment was taken as 12% of the investment in a privately owned utility plant. The following breakdown was used:

Return on money invested	6.0 %
30-year depreciation	1.25
Interim replacements	0.35
Federal income taxes	1.80
Other taxes	2.40
Insurance (excluding liability)	0.20
	12.00%

The bases for the individual charges are discussed in the following paragraphs.

1. Return on Money Invested. The cost of money reflects the downward trend in bond interest since 1960. It is equivalent to financing the plant with one-third equity capital returning 9% after taxes and two-thirds debt capital drawing 4.5% interest.

2. Depreciation. The 30-year depreciation allowance is calculated using the sinking-fund method.

3. Interim Replacements. The sinking fund depreciation allowance does not include replacing capital items that have a shorter life span than 30 years. The 0.35% figure follows FPC practice for evaluating coal-fired plants.

4. Federal Income Taxes. The figure of 1.8% for federal income taxes is based on the assumption that the "sum of the years digits" method of tax deferrals would be allowed. Under such conditions, the percentage of federal income taxes paid on the initial investment the nth year after startup is $0.92 + 0.085n$. Using the sinking-fund method of normalizing this to a constant return per year results in the 1.8% per year charge.

5. Other Taxes. State and local taxes are taken as 2.4% as recommended by the FPC.

6. Insurance. A figure of 0.2% is allowed for property damage insurance following conventional plant practice. This does not include costs of third-party liability insurance.

Appendix B

BASES FOR CAPITAL COST ADJUSTMENTS

Methods used in arriving at the adjusted capital cost for comparison in Tables 6.1, 6.2, and 6.3 of Chapter 6 are discussed in more detail below.

Site Development

All concepts were adjusted as necessary to conform to the conditions at the AEC reference site in Massachusetts. This modification affected primarily the circulating-water systems, which were adjusted, as necessary, to provide a turbine exhaust pressure of 1.5 in. Hg abs at a circulating-water inlet temperature of 57°F.

Land and land rights for all concepts were based on a 1200-acre site costing $360,000. Ground improvements were likewise made common to all concepts on the basis of the PWR estimate of $866,000, since this figure is comparable with other Sargent & Lundy estimates for this site and it does not appear that differences in concepts would affect this item significantly.

Buildings

Variations in building layouts and accounting procedures among the reports that were submitted make it difficult to check reported costs on the basis of reported layouts and building titles. A procedure involving establishment of necessary housing functions and determining that these functions were accounted for was adopted as the most uniformly applicable check of the adequacy of the layouts and uniformity of costs. Although breakdowns by AEC accounts were not available for the HWR and the HTGR plants, some of the buildings were separately accounted for and a rough check of the total cost for Account 212 was obtained for the HWR.

The costs reported for the various housing functions cannot be expected to be comparable because the functions were grouped in different buildings in most cases and the buildings themselves were of widely divergent layouts. The costs of certain buildings could be roughly checked against other estimates for similar buildings or functions. In particular, the turbine buildings, office buildings, and crib houses were comparable with those of other studies. The reported costs were checked against these other estimates and found to be adequately evaluated. An exception was the HTGR turbine building, which is reported to cost about twice as much as the other turbine buildings, but which includes many functions that are separately housed in other concepts.

The building accounts for the HTGR and SGR also include the reactor building, which contains most of the reactor auxiliary systems. The reported estimate for the SGR building is about 3.3 times that for the HTGR

building, but the SGR building has a gross volume of about 2.5 times that of the HTGR building and is subdivided by many shielding walls for numerous auxiliary systems. The reported costs are therefore not inconsistent with the layouts shown. It is noted that although no separate buildings for administration and waste disposal were evident in the design drawings, the reported costs for the SGR included separate estimates of $374,000 and $175,000 for these two buildings. These costs could not be independently estimated, but they are of the same order of magnitude as those reported for similar functions for the PWR and the HWR concepts.

The adjusted costs for the buildings were derived from the following considerations.

Water-Cooled Reactors

1. All plant layouts, with the exception of the reactor building, were taken as similar to the PWR plant layout. Separate buildings were provided for fuel handling and storage, reactor auxiliary equipment, administration, and the turbine-generator plant; the crib house, while separate, was included in Account 232.3. The administration building was assumed to adjoin the turbine building and to house the control room, shops, and laboratories. Some of the waste disposal facilities were considered to be located in the reactor auxiliary building and others in separate structures, whose cost was included under fuel handling and waste disposal. The heating plant building was assumed to be included in the cost of the turbine building.

2. Because of lack of information relative to particular requirements of the water-cooled concepts, other than the PWR, for items such as waste disposal, fuel handling and storage, control room area, and general office and shop areas, it was assumed that the cost of the structures reported for the PWR for these items would be adequate for the other water-cooled concepts. The cost reported for the yard craneway for the PWR was added to the cost of fuel handling in each case.

3. Although some differences are to be expected in the layout of the turbine-generator building (differences arising mostly from the use of different numbers of feedwater heaters and live-steam reheaters), the effect of these differences could not be evaluated without a plant layout and detailed takeoffs. Because the turbines and their auxiliary equipment for the water-cooled plants are all basically similar and because the reported costs are comparable, it was assumed that the turbine-generator building cost for the PWR would be adequate for the other concepts.

Supercritical Steam Plants

1. Building layouts for the reactor, auxiliary equipment, fuel handling, and waste disposal were assumed as reported for each case. All these items are covered by the costs reported, including a cost of $175,000 for a waste-disposal facility for the SGR. The costs of the reactor buildings were not changed from those reported.

2. It was assumed that the turbine-generator plant layout would be similar to that described for the water-cooled reactors, except that a cross-compound turbine would be used. Some increase in the building cost

would be expected as a result, and the reported figure for the SGR is in line with the cost to be expected. Since the HTGR turbine was changed to one equivalent to that of the SGR, the building costs were assumed to be equal for these two concepts.

3. A separate administration building was assumed in each case. For the HTGR this building would house the same facilities as that for the PWR, and its costs would be comparable with that estimated for the PWR. On the other hand, the control room and shops and laboratories are apparently included in the cost of the SGR reactor building; therefore it is reasonable to use the lower figure as reported for the SGR administration building.

Adjusted Costs

Comparison of the adjusted costs with the reported costs indicates a reduction of $510,000 for the PWR as a result of removing the cribhouse from this account. The adjusted cost estimate for the SSCR is equal to that for the PWR but is lower than the reported cost by $1,800,000. Since no details of either costs or layouts were provided for the SSCR, it was not possible to determine the reason for this difference. The estimate for the HWR plants was increased by $631,000, most of which is accounted for in

	Addition	Reduction
Fuel Handling	$127,000	
Reactor auxiliary equipment	599,000	
Office building	213,000	
Waste disposal		$265,000
Turbine building		94,000

Little change was made in the SGR estimate, and the adjusted HTGR total cannot be compared with the reported cost because a total was not reported. The major difference between the HTGR and the SGR estimates lies in the reactor buildings, whose differences are noted above.

All plants were assumed to have stacks of comparable design and therefore of equal cost.

Containment

The original studies of the water-cooled reactors reported containment concepts varying from reinforced-concrete cylindrical structures to spherical steel containers. None of the reports included details of the estimated costs beyond those shown in Table 5.4.

An independent study of the containment requirements was performed in order to normalize all concepts. In this study the following criteria were used:

1. Shielding around the buildings would not be required because the remoteness of the site would provide protection from the radiation effects of fuel meltdown.

2. The containment buildings would contain an instantaneous loss of all coolant in the primary system at operating conditions.

3. The maximum steel plate thickness for on-site fabrication would be 1.5 in.

4. The most economical building design that met the space and containment requirements would be used.

These criteria led to modifications in the designs reported for the water-cooled reactors. The cost of the buildings was estimated from a takeoff of major cost items as noted below:

	Unit Cost
Excavation	
Earth	$1.50 per yard
Rock	$17.50 per yard
Backfill	$1.60 per yard
Disposal	$0.50 per yard
Dewatering	$30,000
Structures	
Reinforcing steel	$240 per ton
Plate steel for shell	$1000 per ton
Concrete for shell	$50 per yard
Substructure, concrete	$65 per yard
Concrete for floors, walls, and shields	$95 per yard
External insulation	$3 per ft^2
Coating and painting	$4 per ft^2
Air lock	$100,000
Miscellaneous steel	$160,000
Building services	
Plumbing and drainage; Heating, cooling, and ventilation; Lighting; Fire protection	$110,000

As a result of these studies it was found that the estimated cost of the buildings as originally designed was within 10% of the reported cost. Takeoffs from the modified buildings were used with the foregoing unit costs to obtain the normalized estimates. It is to be noted that the cost of the PWR containment structure was increased by about 8%, while that of the HWR was decreased by nearly 30%, with the range from the highest cost structure (PWR at $\$7.7 \times 10^6$) to the lowest (SSCR at $\$5 \times 10^6$) representing a range of $\$2.7 \times 10^6$ or 35% of the PWR building cost. A major reason for the differences in cost is the difference in coolant volume and heat content of the various concepts. These quantities determine the building volume or design pressure, subject to the condition that the building has sufficient diameter to house the reactor primary system.

An estimate of the incremental cost of providing a steel-lined reinforced-concrete containment building for the HTGR was also prepared, since, as explained in Chapter 6, the addition of secondary containment per the HTGR alternate design was assumed to be required. The building layout and design pressure were based on those developed by General Atomic: a flat-bottomed 135-ft-diam cylinder with a hemispherical dome and a design pressure of 29 psig. The estimated additional direct cost represented by this structure, excluding all internal structures, is $2,400,000, or about $2.40 per kilowatt of electrical capacity. It adds $3,300,000 to the total investment for this plant.

Reactor Vessel

Two of the water-cooled reactors, the PWR and the SSCR, have carbon steel pressure vessels internally clad with stainless steel to house the core. The heavy-water reactors use "cold" D_2O in a calandria tank and the D_2O coolant is contained at operating pressure in Zr–2.5% Nb alloy pressure tubes that pass vertically through the calandria. The SGR also employs a calandria design to separate the sodium coolant and graphite moderator. The HTGR is contained completely within a prestressed-concrete pressure vessel. The concrete vessel is used to contain the steam generators and helium circulators as well.

The contractor's reported costs in all cases, except the SSCR, state the estimated installed cost of the reactor vessel. The SSCR vessel is included in the single value presented for Account 221. Most recent information on the PWR indicates that the reactor vessel now being considered is somewhat smaller than that described in ref. 1 of Chapter 3 and therefore should be adjusted accordingly. Therefore the unit costs of the shop-fabricated cylindrical pressure vessel, including the head and internals, were derived using the vessel described in ref. 1 of Chapter 3, and then the unit costs were used to determine a revised vessel and internals cost corresponding to the most recently reported vessel and head weight of 591 tons.

A single cost for all of Account 221 was furnished by B & W for the SSCR. No data on which to base a breakdown of the cost was available; however, using the cost information given for the PWR as a basis, it appears that the $9,213,000 estimate includes an appropriate amount for the reactor vessel. Although a vessel weight of 313 tons was given for the SSCR, little is known of the control rod and piping connections, all of which affect cost.

The costs for the two heavy-water reactors (HWR-U and HWR-Th), as shown in the references of Chapter 6, are broken down into their major component parts; for example, calandria, coolant tubes, headers, etc. The bases for compiling these cost data were reported by Du Pont (see ref. 13, Chap. 6) and appear to be adequate.

Costs for the HTGR prestressed-concrete reactor vessel were compiled by General Atomic on the basis of a thorough feasibility analysis, together with supplier's estimates of material costs. Vessels similar to this one are currently being constructed in both France and Great Britain on the basis of known technology. No detailed breakdown of the GA estimate of

the reactor vessel, internals, and in-core instrumentation costs was available from which the reported costs could be verified. However, based on ORNL estimates of a similar concept, it appears that about $\$3 \times 10^6$ should be added to the estimate ($\$1.6 \times 10^6$ for the vessel and internal structures; $\$1.4 \times 10^6$ for in-core instrumentation). This was done in arriving at the adjusted Account 22 cost.

The SGR cost data are reported for each component of the core structure and, on the basis of reported data, appear to be adequate.

Reactor Controls

All the reactors use rod control systems that are tailored to match their particular requirements. The 109 control rods and drives for the PWR are estimated by Westinghouse to cost approximately $47,000 each, which is consistent with costs reported elsewhere. Using this as a basis, the unit cost of the control rods and drives for the remaining reactors appear to be adequate and can be used as reported by the suppliers.

Both the PWR and HWR use a liquid poison system for either reactivity control or safety. The costs for the appurtenances of this system were available for only the HWR, and it is assumed that this system was included in Account 221.21 for the PWR.

The data furnished on the SSCR do not permit an analysis of the B & W estimate for reactor control equipment. However, using the Westinghouse data for the PWR, it appears that such equipment has been adequately covered in Account 221.

Reactor Auxiliary Cooling and Heating System

The PWR has $848,500 worth of equipment to remove decay heat from the primary loop and to act as an auxiliary heat sink during emergencies. Using this as a basis, it is not clear that the HWR plant has a comparable system, since its reported cost is only $360,000. It was decided therefore to use the PWR estimates for the adjusted costs so that both systems are comparable. It cannot be determined from reported information whether the SSCR cost for Account 221 includes a system comparable with that of the PWR. However, since a rough breakdown of the B & W estimate, based on the PWR costs, shows substantial cost margins, it was decided to assume that this system was included.

Reactor Plant Crane

Except for the HTGR (which uses no reactor building crane) each of the plants is furnished with an appropriately sized reactor building crane. The size of the cranes varies considerably among the plants, depending upon the heaviest piece to be lifted, which leads to a corresponding variation in the estimated costs. In each case the manufacturer's costs were used.

Heat Transfer Systems

The reported costs were adjusted on the basis of technical data provided by the manufacturer coupled with the reviewer's judgement of what these costs might be on a normalized basis. The costs reported by Westinghouse for Account 222 were examined and found to be consistent with costs reported elsewhere. On this basis it was decided to use the PWR estimates as a basis for judging the costs of the water-cooled reactors. One exception to be noted is that Westinghouse based the PWR on the use of canned-motor pumps, while the other water-cooled reactors included controlled-leakage pumps. Since issuing the PWR design report, Westinghouse has also converted to using controlled-leakage pumps, so their costs for Account 222.1 were modified accordingly. In the size and pressure range being considered, canned pumps cost about $13/gpm, while equivalent controlled-leakage pumps cost about $8.00/gpm. Since the cost for the pumps alone was not stated by Westinghouse, it was decided to apply a differential of $5.00/gpm to the primary coolant flow rate and reduce the PWR Account 222.1 by a corresponding amount. Since the controlled-leakage pumps require certain auxiliary equipment not needed with canned pumps, such as seal-injection pumps, collection tanks, piping, etc., an amount of $135,000 was reinserted in Account 222.1 for the PWR. Thus Account 222.1 for the PWR was reduced by $2,000,000.

The SSCR costs, as reported by B & W, were given as a total for Account 222 only. By separate transmittal, B & W furnished prices on the primary circulating pumps, piping, and steam generators. A technical analysis of the steam generators showed that the heat transfer coefficient used by B & W was approximately 1.5 times that used for the PWR. From the information given on temperatures and flow rates, it was difficult to see how such a variation could occur. Therefore, the costs for the SSCR steam generators, pumps, and piping were modified to correspond with those of the other water-cooled reactors.

The costs reported by Du Pont for the HWR piping and pumps were examined and appeared to be consistent with those reported by Westinghouse after they were adjusted to the use of controlled-leakage pumps. On this basis it was decided to use the Du Pont estimates.

The HTGR and SGR cost estimates were examined similarly. Since the primary loop equipment for these two plants is based on completely different designs, it is difficult to determine on the basis of the reported data any common basis for comparison. However, the costs appear to be reasonable compared with those of the other plants, so it was decided to use these costs as reported. The reported costs for the HTGR heat transfer system included electrical and instrumentation cost allowances. These allowances were distributed to the appropriate accounts in arriving at the adjusted costs.

Nuclear Fuel-Handling and Storage Equipment

In general, the reported costs for fuel-handling machinery remain unchanged. In view of the special nature of this equipment and the fact

that in each plant the machinery must be tailored to perform specific functions not common among the plants, it was judged most appropriate to accept the reported costs. In the case of the HWR and HTGR, substantial sums were included for off-site shipping casks or cars or both. These costs were deducted from these accounts for consistency.

Only one of the plants, the thorium-fueled heavy-water plant (HWR-Th), is designed for on-power refueling. A sum of $5,000,000 is included in reported costs for this refueling machine. A substantial development effort on an on-power refueling machine is currently under way in the U.S., and Canada has similar machines in operation at NPD-2 and at Chalk River. It is felt that for purposes of this study the $5,000,000 estimate by Du Pont for such a machine is probably adequate, exclusive of research and development costs. It should be recognized, however, that the latter is likely to be substantial and that little information is available at present to substantiate the cost estimate for the machine.

Radioactive Waste Treatment

The reported costs for radioactive waste disposal range from $371,000 to $400,000, with the exception of the SSCR plant, for which $800,000 is estimated for waste-treating equipment. Although it is not stated, the SSCR estimate apparently includes some of the tankage and equipment required because of the varying concentration of D_2O in the coolant.

Since only a small variation occurs among the plant costs estimated for waste treatment, it was judged appropriate to apply a uniform cost of $400,000 for the system, with the exception of the SSCR. This procedure should eliminate any difference between the PWR, for which $112,000 was reported for shielding structure, and the other plants. All shielding and other housing costs for the system are covered in the building accounts. In view of the lack of detail for the SSCR system, it was decided to adopt the designer's estimate for the waste disposal system, since it is probable that this system would be substantially more costly than the others because of the complexities involved in varying the coolant composition.

Instrumentation and Control

The most complete breakdowns of the instrumentation and control account were provided for the PWR and the SGR. A less definitive breakdown appears for the HWR, and essentially no data were provided for the HTGR and SSCR.

The estimates reported by Westinghouse, Du Pont, and Atomics International were judged to be adequate. They appeared to include all elements of the plant control system. These reported costs were therefore left unchanged. In the case of the PWR, an amount of $335,000 was relocated from Account 242.1 to Account 226.3 to correspond more nearly to the other account breakdowns.

Since no breakdown for Account 226 was given for the SSCR, it was difficult to judge how to distribute the reported $1,314,500. It was

assumed that all but the reactor control and in-core instrumentation should be about equal to that reported for the PWR, and the costs were adjusted accordingly. Presumably the requisite in-core instrumentation for the SSCR is less extensive than for the PWR, but its cost would be partially offset by the cost of instrumenting the D_2O concentration control system. On this basis an amount of $1,500,000 was included for SSCR Account 226.1.

The costs for HTGR Account 226 were supplied by General Atomic and were combined with cost data compiled by Sargent & Lundy during earlier work on a concept similar to this one. In view of this the HTGR costs reported by GA were used.

For consistency, an amount of $1,000,000 reported by Atomics International for the SGR control valves was relocated in Account 228.

Feedwater Supply and Treatment

The total cost for the feedwater supply and treatment system for the PWR was accepted as reported after examining the individual costs for the equipment included. The adjusted costs for the feedwater heaters were based on the heat duty of each heater stage. Each of the six stages of feedwater heating has three feedwater heaters. Because the thermal cycle for the HWR requires a lower final feedwater temperature than the PWR, it has one less stage of feedwater heating in the low-pressure heaters. The estimated cost of the feedwater heaters for the poison-controlled PWR was subtracted from the reported cost of feedwater heaters and reheaters and the remainder was added to the total cost as item 227.6 — Reheaters.

The HTGR has seven stages of feedwater heating with two feedwater heaters per stage. The SGR has a higher feedwater flow and eight stages of feedwater heating, with two feedwater heaters per stage. The feedwater heater costs for the HTGR and the SGR were based on heat duties. Although the HTGR and SGR have fewer feedwater heaters than the PWR and HWR, their design pressures are higher and therefore the total costs are similar for all the reactor concepts.

The feedwater makeup and purification systems for the PWR and HWR are based on using a flash evaporator, while the HTGR and SGR use full-flow condensate polishing demineralizers because of the once-through boiler design of their steam generators. These differences in design are reflected in adjusted costs of these systems. The reported costs for the feedwater pumps and drives for the poison-controlled PWR were used for the SSCR after estimating the costs of these pumps based on the reported design conditions.

The HWR feedwater pumps were estimated on the basis of three pumps, the same number as used for the PWR. The difference between the HWR feedwater pumps and the PWR feedwater pumps is the use of fluid couplings for pump control; other than this the pumps of the PWR and HWR have nearly the same design flows, with the design discharge head of the HWR feedwater pumps being approximately 100 psi lower.

The HTGR uses one feedwater pump driven by the main turbine shaft and having capacity for 75% of the required flow and two motor-driven feedwater pumps, each with capacity for 12 1/2% of the full flow. The adjusted cost also includes a booster pump for the 75% capacity feedwater pump

driven by the low-pressure turbine shaft and a motor-driven shutdown pump. The SGR uses two one-half capacity turbine-driven feedwater pumps. Cost estimates for these pumps were adjusted by Sargent & Lundy.

Steam, Condensate, and Feedwater Piping

The total cost reported for the PWR for steam, condensate, and feedwater piping was accepted on the basis of the design data available and Sargent & Lundy's experience in estimating costs for similar systems. The cost reported for the SSCR was determined to be adequate based on the fact that there are five steam generators as compared with seven steam generator loops for the poison-controlled PWR, and this could account for a slightly lower cost in piping. With the limited amount of data available, it was not practical to attempt to reestimate the piping costs for this concept.

For the HWR, steam, condensate, and feedwater piping costs were estimated from Sargent & Lundy's experience in the design of this concept. The lower cost, as compared with the poison-controlled PWR cost, is accounted for by the smaller number of feedwater heaters and the fact that it is a nonreheat cycle.

The HTGR cost for this account was adjusted to agree with Sargent & Lundy design data, equipment takeoffs, and cost estimating performed on this reactor concept.

The steam, condensate, and feedwater piping costs for the SGR were adjusted based on Sargent & Lundy's experience on the HTGR. The estimated cost for this system should be less than the cost for the HTGR because of the shorter lengths of piping from the steam generators to the turbine generator.

Other Reactor Plant Equipment

The facilities and equipment for maintenance of radioactive equipment, portable shielding, storage tanks and other miscellaneous appurtenances are included in the "other reactor plant equipment" account. Substantial variations among the reported costs exist for this account. Since essentially no detail of what was included here was available, it was decided to adopt the reported values. These accounts for the HTGR and SGR are notable, being $720,000 and $687,000, respectively. However, each of these plants has radioactive decontamination systems suitable for decontaminating major equipment for maintenance and therefore would be expected to be more costly than the others.

Turbine-Generator Plant

The cost of the turbine-generator for the PWR was compared with the manufacturer's information on this size and type of turbine-generator, some of which was contradictory, and was accepted as reported and used

as a basis for the cost of the turbine-generator of the SSCR, which was assumed to have similar thermal conditions and would therefore use the same turbine-generator. The turbine-generator cost for the HWR was based on a nonreheat, six-flow, tandem, compound turbine-generator with 43-in. last stage blades. Because of the uncertainty in cost estimates for turbines in this class, the reported cost for the PWR turbine was used. A multiplier of 0.78 was applied to reflect current pricing policies.

The pricing of the turbine-generators for the HTGR and SGR was based on current book prices and, also, a multiplier of 0.78. The HTGR turbine was changed to a four-flow, cross-compound, 43-in.-blade unit, which is the same as the unit used for the SGR. The difference in price for these two units reflects the higher steam temperatures of the HTGR.

An adjustment in the turbine foundation prices for all concepts was made based on Sargent & Lundy's experience in the design of 1000-Mw(e) HWR plants using tandem-compound turbine-generators and 1000-Mw(e) molten-salt converter plant using cross-compound turbine-generators.

Circulating-Water Systems

An evaluation of the design conditions of the circulating-water systems for the PWR and HWR, based on the AEC hypothetical site, showed that the circulating-water requirements for these plants would be nearly the same and similar equipment sizes would be used. The only difference occurs in the HWR circulating-water pumps, which have a slightly higher flow. The pricing of the pumps, traveling screens, circulating-water lines, and intake and discharge lines was based on estimates of similar systems described in the references of Chapter 6.

The reported design condenser pressure of the HTGR was 1 in. Hg abs. For uniformity, this was changed to 1.5 in. Hg abs. Based on the AEC site, the circulating-water requirements become identical to those of the SGR. The intake and discharge structures were priced according to Sargent & Lundy estimates of equivalent structures located on the AEC hypothetical site.

Condensers and Auxiliary Equipment

The costs reported for the condensers and their auxiliary equipment were not subdivided in accordance with the AEC accounting system. By using equipment data reported for the poison-controlled PWR, condensers and condensate pump costs were derived by Sargent & Lundy's estimating procedures. These costs were used for the three water-cooled reactor concepts. The condensers and condensate pumps for the HWR were sized and priced on the same basis as those of the PWR. The HWR condensers have a larger surface area than the PWR condensers, while the latter have deaerating hotwells.

The condensers for the HTGR were redesigned for a 1.5-in.-Hg exhaust pressure and circulating-water temperatures corresponding to the AEC hypothetical site, as described above. The condenser surface area for the HTGR

is slightly smaller than that for the SGR, while the condensate flows and heads for the two concepts are similar enough so that the same condensate pumps can be used. The adjusted costs for the condensers and condensate pumps for the HTGR and the SGR are based on Sargent & Lundy's estimates.

Accounts 234, 235, 236, 237, 238, and 239 were all assumed to be equal for similar turbine plants. Therefore the costs in these accounts for the pressurized-water and the heavy-water reactors were taken from those reported for the PWR, and the costs for the HTGR were made equal to those costs reported for the SGR.

The costs as reported for turbine plant boards, instruments, and controls for the HTGR and SGR were adjusted to agree with the costs reported for the PWR concept by placing the control valve costs in Account 228, Steam, Condensate, and Feedwater Piping.

Electrical Systems

Only the report on the PWR included a diagram of the electrical system for the plant. No design description of the electrical systems for the other concepts was available, with the exception of the HTGR, for which some data pertaining to transformer capacity and switchgear arrangement were obtained.

The design normalization consisted of analyzing the systems of the PWR and the HTGR and using these as a standard for the water-cooled reactors and the SGR, respectively. An estimate of necessary transformer capacity and switchgear requirements for the unreported systems was obtained by adding or subtracting feeder breakers and transformer kva as required by the differences in numbers and kva requirements of major motors. This procedure resulted in the following:

Concept	Estimated Auxiliary Power (kva)	Number of 4160-v Feeder ABC's
PWR	75,000	45
SSCR	70,000	43
HWR	85,000	44
HTGR	37,500	34
SGR	48,000	42

It was observed from estimates supplied by the designers that only the estimates for the PWR and SGR were prepared in accordance with the AEC Classification of Accounts and that the variation in estimated costs for those items that should be equivalent could not be justified. It is apparent that the accounting procedure was not uniformly interpreted in these two cases. The reports on the remaining concepts either contained no breakdown or provided a breakdown that was not in accordance with the AEC classification.

In normalizing the costs, it was assumed that the system described for the PWR was adequate, with the foregoing exceptions, for the other

water-cooled reactors. It was further assumed that the HTGR electrical system was adequate, again with modifications as noted above, for the SGR. Current manufacturer's catalog prices were obtained for the isolated-phase bus duct and the auxiliary transformers and switchgear, and the labor costs were estimated by Sargent & Lundy.

The remaining items, including conduit, cable, battery systems, duct runs, cable pans, and local controls, were estimated on a gross basis using Sargent & Lundy's construction records for large single unit stations. These records indicate a price range from $2 to $3 per kilowatt of capacity for units of up to 530 Mw. In view of the larger capacity of the reactors discussed here, the unit cost was assumed to be $2.50 per kilowatt for those systems with all-motor drives (the water-cooled reactors) and $2.00 per kilowatt for the HTGR and SGR, which use turbine drives for the largest pumps.

Miscellaneous

Those items classified as miscellaneous under Account 25 include cranes and hoists, compressors and vacuum systems for general station use, and furniture, fixtures, etc. for general station use. All concepts should have similar costs for these items, since their magnitude, other than that of the turbine room crane, is not dependent on the type of plant being erected. A figure of $800,000 was used to cover this account. This amount is representative of that used in other studies of 1000-Mw plants and corresponds approximately to the figures reported by the designers.

Appendix C

FUEL FABRICATION FLOWSHEETS

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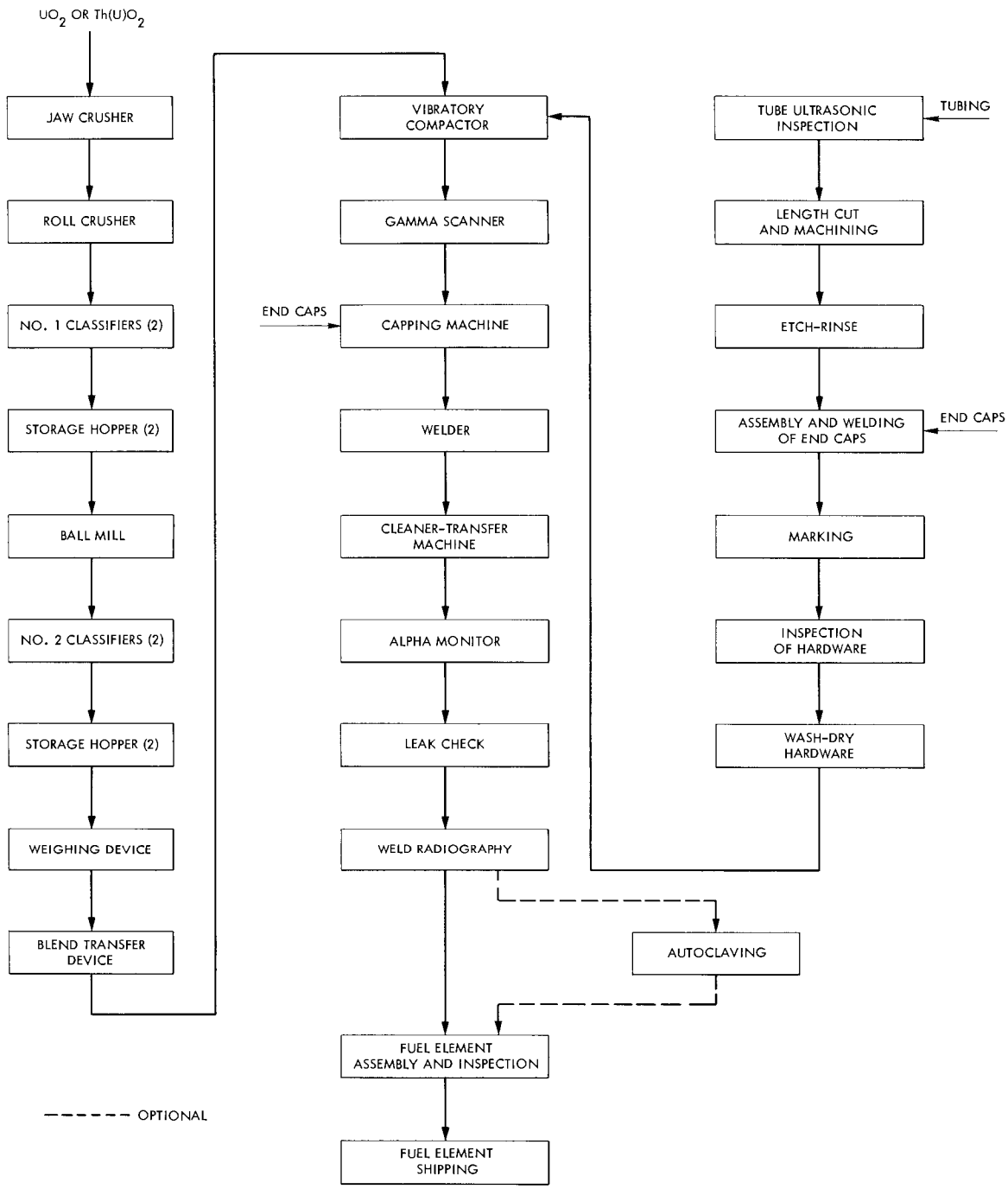

Fig. C.1. Fabrication Flowsheet for Vibratory Compaction of Fuel in Rod Bundles.

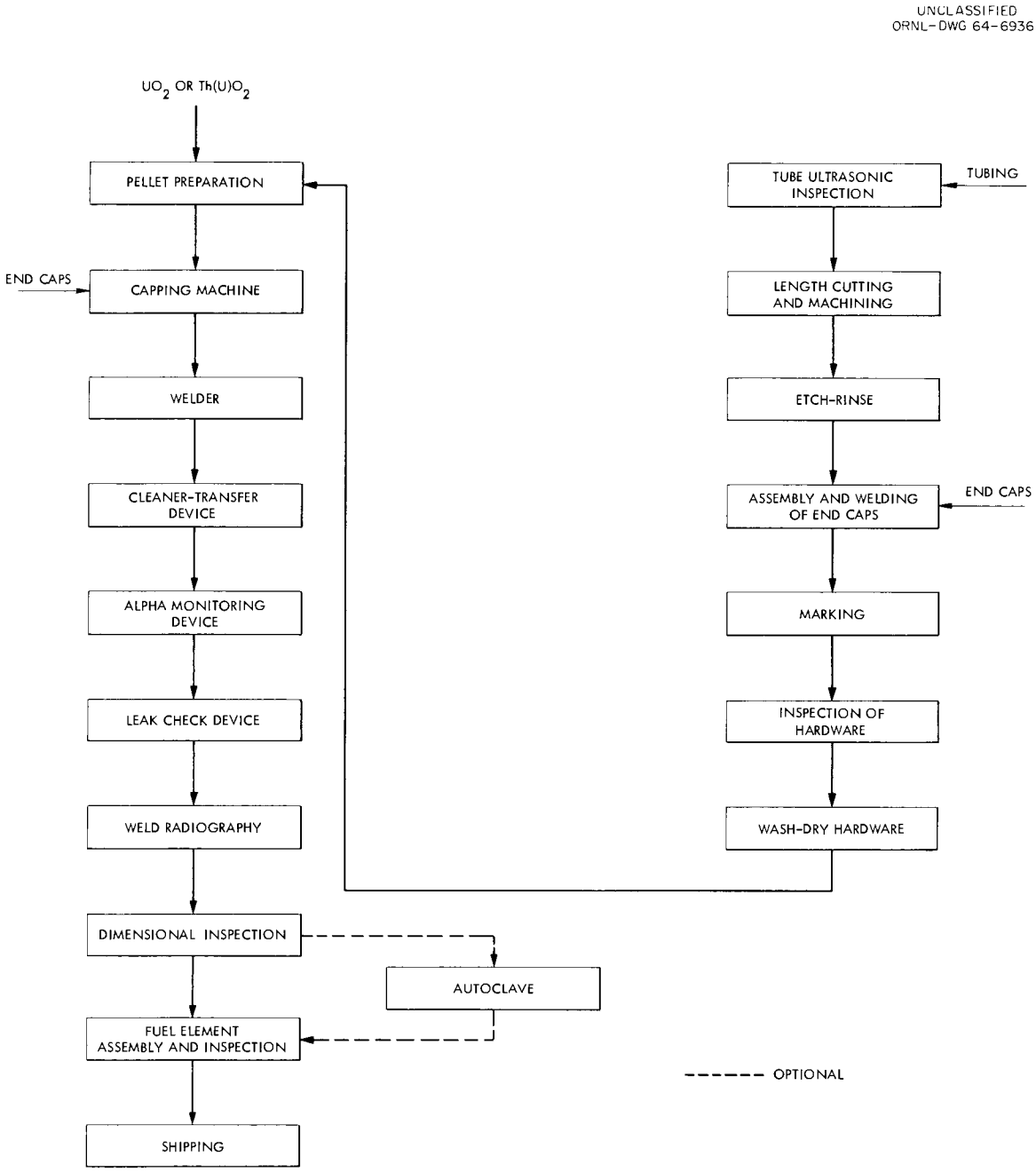

Fig. C.2. Fabrication Flowsheet for Rod Bundles Containing Pelletized Fuel.

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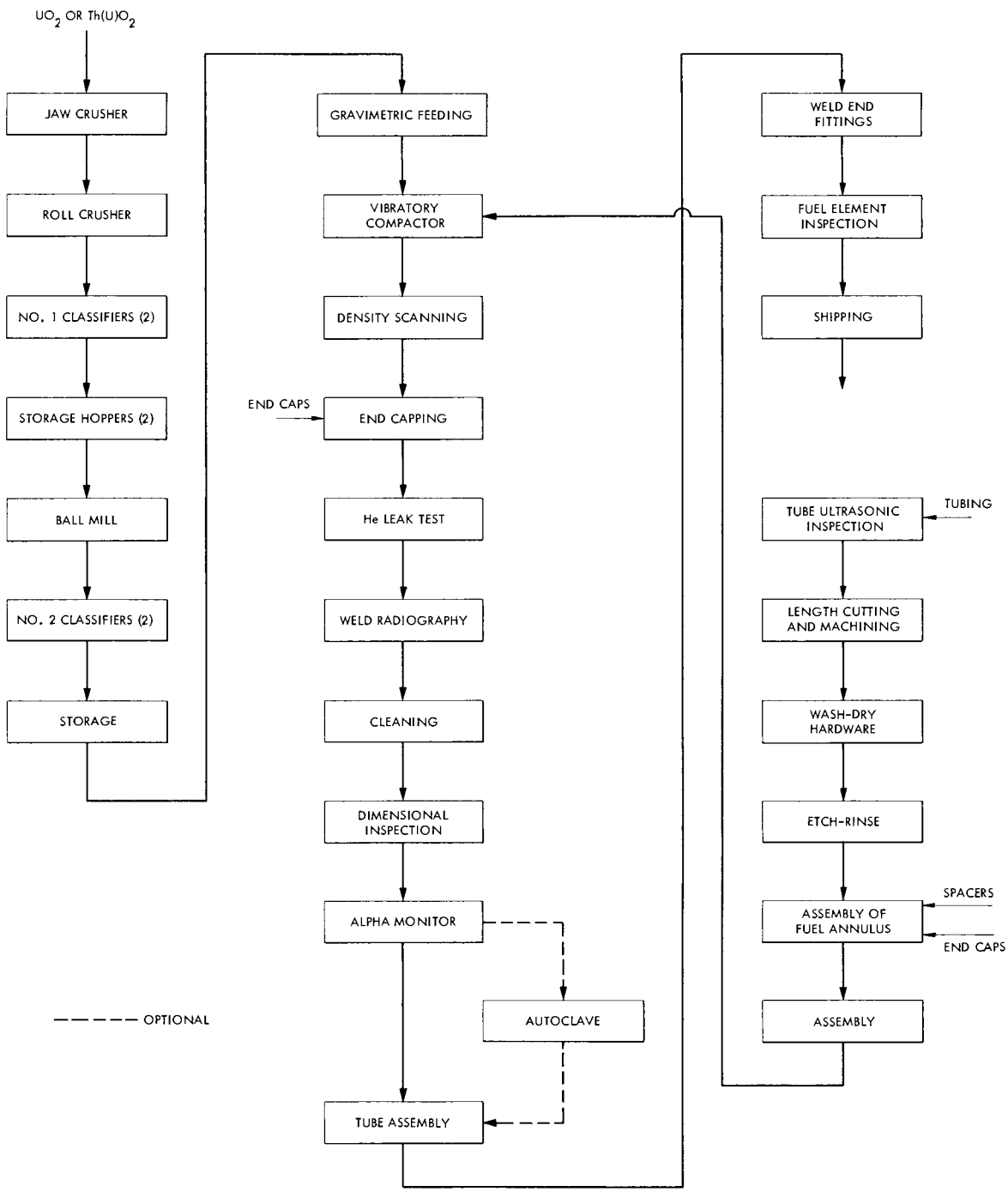

Fig. C.3. Fabrication Flowsheet for Vibratory Compaction of Fuel in Nested Tubes.

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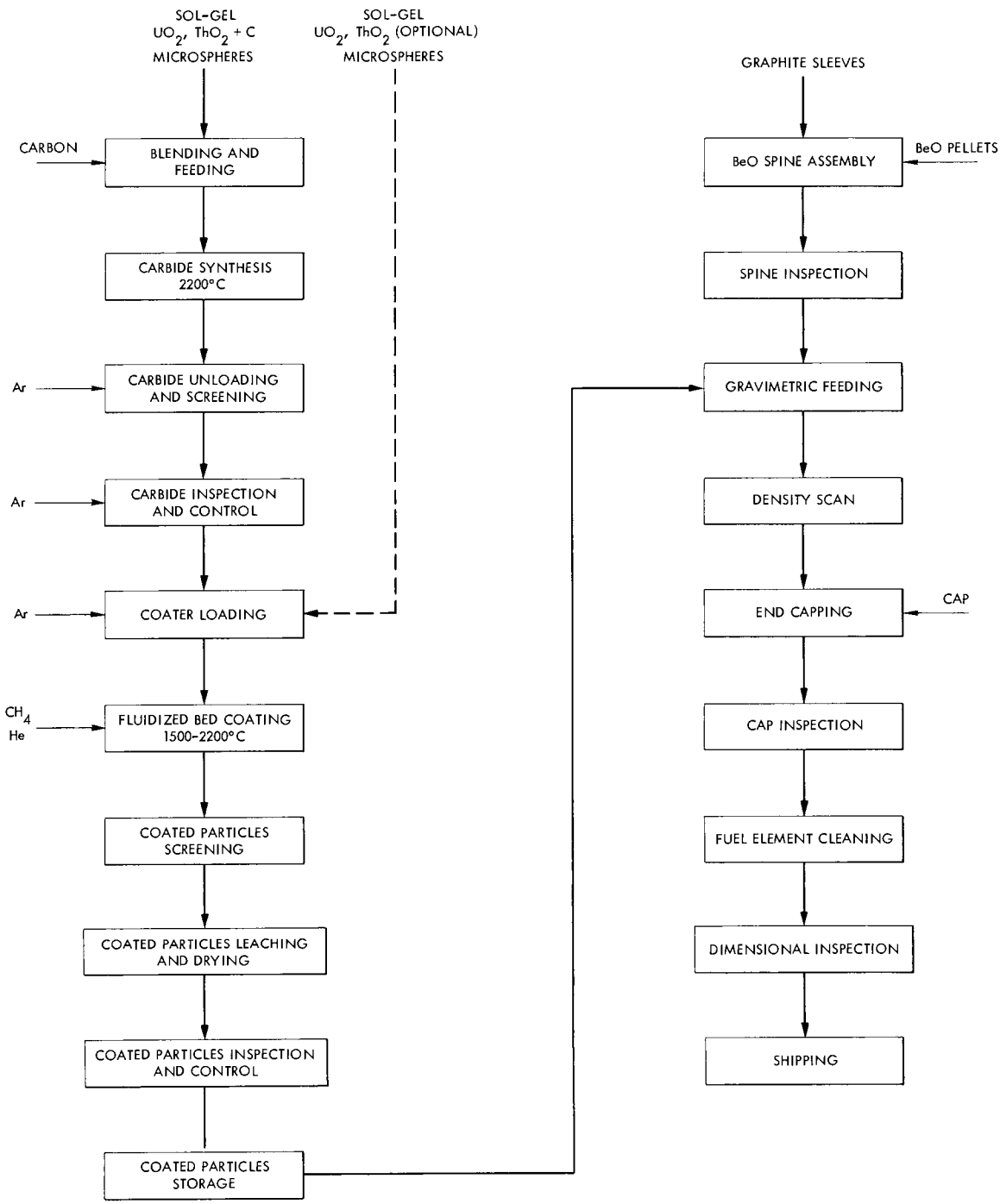

Fig. C.4. Fabrication Flowsheet for HTGR Fuel.

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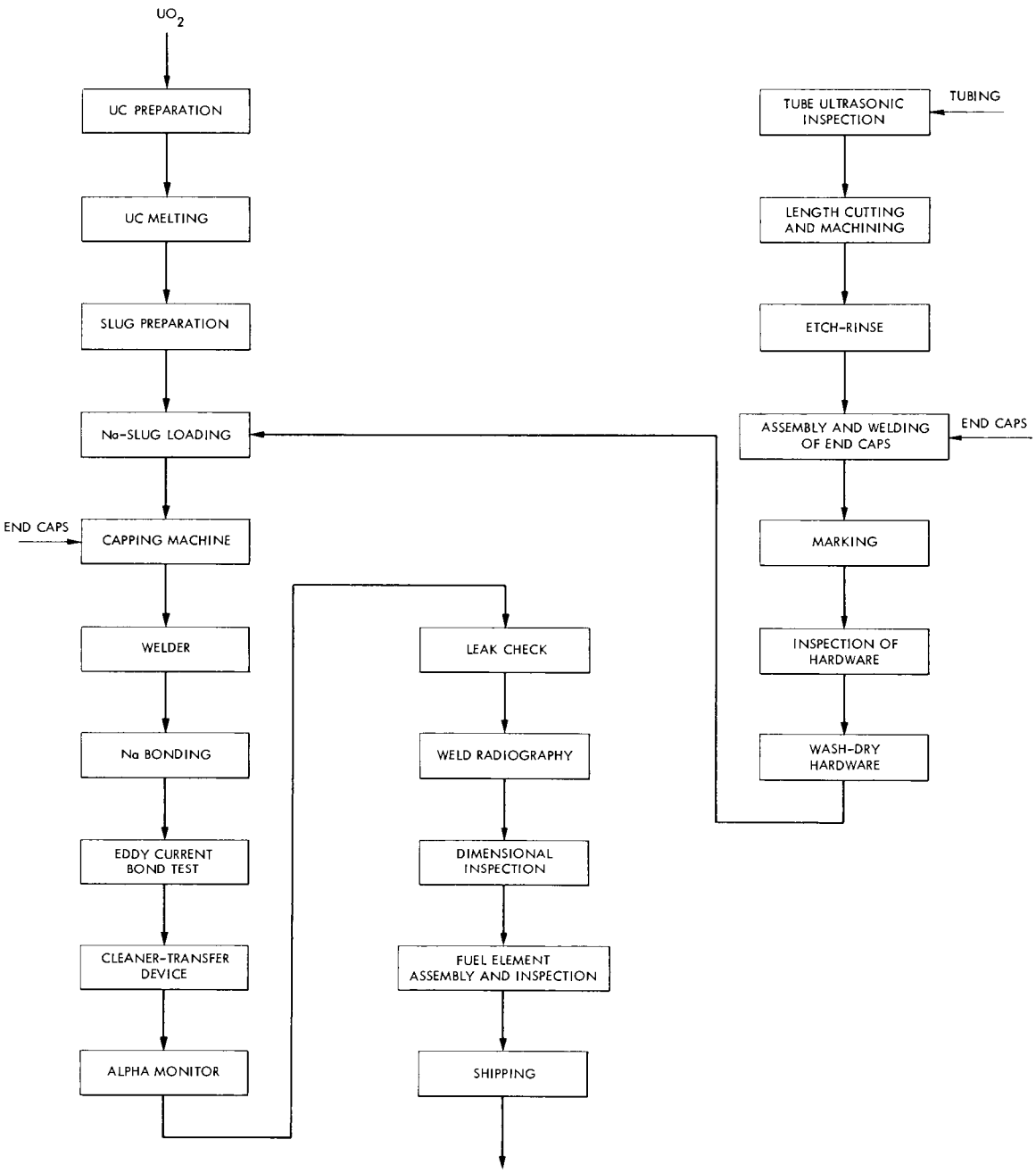

Fig. C.5. Fabrication Flowsheet for SGR Fuel.

Appendix D

SUPPLEMENTARY INFORMATION ON FUEL PROCESSING COSTS

Table D.1. Summary of Single-Purpose Processing Plant Cost Estimates

	PWR		SSCR		HWR-U		HWR-Th		HTGR					SGR	
									Full Recycle			Discard	Recover		
	Case 7	Case 9	Case 65	Case 70	Case E-2	Case E-4	Case A-1	Case A-4	Case 8	Case 9	Case 1	Case 10	Case 10	Case 1	Case 5
Reference material burnup, Mwd/kg	21.25	32.34	29.58	42.53	12.54	17.54	23.0	39.6	44.1	58.0	75.0	48.5	48.5	16.75	33.75
Thermal efficiency, %	31.1	31.1	31.2	31.2	26.8	26.8	26.1	26.1	44.4	44.4	44.4	44.4	44.4	43.6	43.6
Throughput rate, MT/year	663	435	475	330	1300	932	730	424	227	172	133	206	206	600	298
Processing days per year	156	210	255	273	155	155	155	155	155	155	175	155	155	155	155
Processing rate, MT/day	4.23	2.08	1.86	1.21	8.40	6.01	4.71	2.73	1.46	1.11	0.760	1.28	1.37	3.87	1.92
Construction cost, 10^6 $	54.9	49.4	55.6	52.2	61.6	58.1	63.1	57.1	51.5	49.3	46.8	48.7	54.4	55.4	49.7
Land, startup, and working capital, 10^6 $	3.6	3.3	3.7	3.0	4.2	4.0	4.4	4.0	3.3	3.2	3.0	3.1	3.5	4.6	3.2
Total capital investment, 10^6 $	58.5	52.7	59.3	55.6	65.8	62.1	67.5	61.1	54.8	52.5	49.8	51.8	57.9	59.0	52.9
Annual operating cost, 10^6 $	5.37	4.63	5.47	5.00	6.33	5.83	6.55	5.69	4.90	4.62	4.29	4.53	5.30	5.45	4.69
Annual ultimate waste disposal charge, 10^6 $	2.52	2.52	2.51	2.51	2.75	2.75	2.79	2.79	2.05	2.05	2.05	2.05	2.05	2.05	2.05
Total cost, 10^6 $/year															
At 15% fixed charge	16.7	15.1	16.9	15.9	18.9	17.9	19.5	17.6	15.2	14.5	13.8	14.3	16.0	16.3	14.7
At 22% fixed charge	20.8	18.7	21.0	19.7	23.6	22.2	24.2	21.9	19.0	18.2	17.3	18.0	20.1	20.5	18.4
At 30% fixed charge	25.4	23.0	25.8	24.2	28.8	27.2	29.6	26.8	23.4	22.4	21.3	22.1	24.7	25.2	22.6
Unit cost, $/kg															
At 15% fixed charge	25.1	34.6	35.5	48.0	14.5	19.2	26.7	41.6	66.8	84.5	104	69.6	77.8	27.2	49.2
At 22% fixed charge	31.3	43.0	44.3	59.8	18.1	23.9	33.2	53.3	83.8	106	130	87.2	97.6	34.1	61.6
At 30% fixed charge	38.4	52.7	54.3	73.3	22.1	29.2	40.6	63.2	103	130	160	107	120	42.0	75.8
Unit cost, mills/kwhr(e)															
At 15% fixed charge	0.159	0.144	0.161	0.151	0.180	0.170	0.185	0.167	0.144	0.138	0.131	0.136	0.152	0.155	0.140
At 22% fixed charge	0.198	0.178	0.200	0.187	0.224	0.211	0.230	0.208	0.181	0.173	0.164	0.171	0.191	0.195	0.175
At 30% fixed charge	0.242	0.219	0.245	0.230	0.274	0.258	0.282	0.255	0.223	0.213	0.202	0.210	0.235	0.240	0.215

Table D.2. Summary of Dual-Purpose Processing Plant Cost Estimates

Basis: 10,000-Mw(e) PWR plant plus 10,000-Mw(e) second fuel plant as indicated; costs in PWR column are for a 20,000-Mw(e) single-purpose PWR plant

	PWR	SSCR	HWR-U	HWR-Th	HTGR	SGR
Individual fuel load						
Burnup, Mwd/kg	25.00	30.00	15.40	22.00	51.40	35.00
Thermal efficiency, %	31.1	31.2	26.8	26.8	44.4	43.6
Throughput rate, MT/year (for 10,000 Mw)	376	312	708	522	124	191
Combined fuel load						
Operating days per year	155	194	155	155	171	174
Nominal processing rate, MT/day	4.85	4.64	7.01	8.05	3.42	3.27
Construction cost, 10^6 $	57.4	59.5	61.3	63.8	59.7	55.6
Land, startup, and working capital, 10^6 $	4.0	4.2	4.3	4.5	4.0	3.9
Total capital investment, 10^6 $	61.4	63.7	65.6	68.3	63.7	59.5
Annual operating cost, 10^6 $	5.72	6.03	6.29	6.66	6.05	5.66
Annual ultimate waste disposal charge, 10^6 $	2.99	2.99	3.14	3.14	2.74	2.75
Total cost, 10^6 $/year						
At 15% fixed charge	17.9	18.6	19.3	20.0	18.3	17.3
At 22% fixed charge	22.2	23.0	23.9	24.8	22.8	21.5
At 30% fixed charge	27.1	28.1	29.1	30.3	27.9	26.3
Unit cost, mills/kwhr(e)						
At 15% fixed charge	0.128	0.133	0.138	0.143	0.131	0.123
At 22% fixed charge	0.158	0.164	0.170	0.177	0.162	0.153
At 30% fixed charge	0.193	0.200	0.208	0.216	0.199	0.188

Appendix E

COMPUTER PROGRAMS

D. R. Vondy

Computer Code TONG for Point Depletion Calculations

Determination of the 30-year history of a reactor core is a complex problem, especially if the core is only partially refueled at the end of each cycle and fuel is reprocessed and recycled. For example, to determine such a history for the HTGR concept with an eight-neutron-group treatment in two dimensions would require several days of IBM-7090 time, and the accounting problem would not be solved easily. To obtain consistent calculations for a number of reactors makes a single computer program desirable. The program should be fast in order to reduce computing costs in a comparative study.

The simplest geometric approximation is consideration of one point in space. The point calculation assumes that exposure of a sample of material (which represents a core average) to a flux at the level necessary to develop the average power density, will approximate the spatially averaged time behavior of the actual core. Such a calculation can be reliable only when the core is large and reflected.

Spatial effects are important, and some consideration must be given to the influence of such effects on reactor performance. Generally, the flux peaks toward the core center and causes high depletion there, as well as high importance of fission products. Fuel near the edge or end of a reactor receives low exposure and thus has low utilization. To minimize the cost of power generation generally requires a high average power density and a high average exposure of the fuel. Thus schemes are devised to flatten the power with techniques such as varying lattice dimensions with position, varying loading of fuel or fertile material or both, and fuel-management procedures. Neutron loss to the reflector is increased by power flattening; such loss may be approximated in a point calculation by use of an effective buckling normalized to the actual leakage.

The IBM-7090 computer code TONG was developed at ORNL to determine a core history over many cycles using the single point, multineutron-group model. Broad-group microscopic cross sections are calculated from a fine-group library. The history is established and the fuel cycle cost is calculated in a single machine run. The fast memory of the computer is filled with different parts of the program and data about ten times in a normal calculation. Typically, an above-thermal-energy calculation is made in the P_1 approximation considering 62 energy groups using both group cross sections and resonance parameters with shielding; this calculation is a modified version of the code GAM-I developed at General Atomic. The cross sections for the thermal groups are obtained from a 30-energy-group, one-dimensional, cylindrical cell calculation made by an integral approximation to transport theory. This calculation is a modified version of the code THERMOS developed at Brookhaven.

A representative 92-group, lethargy flux spectrum is shown in Fig. E.1. A problem of coupling between the thermal- and fast-range spectrums may be noted from the discontinuities between the GAM and THERMOS spectra. This problem is maximized in a situation of high supression of flux in a fuel zone when the input nuclide concentrations are such that the system is very far from being critical.

Broad-group, microscopic cross sections are obtained by weighting the fine-group cross sections over the calculated flux spectrum, and these are used for depletion calculations with TONG, as well as by other codes employed at ORNL. In a typical depletion problem, initial nuclide concentrations are specified in each of several fuel batches. In TONG these concentrations are averaged, and the point eigenvalue problem is solved to establish the fluxes. Depletion history is followed for each fuel batch exposed to the single set of fluxes, which are recalculated at specified time intervals.

To begin a cycle, the fuel loading in a newly loaded batch may be specified or a desired multiplication factor, k, may be satisfied (usually without control rods) by adjustment of one or more nuclide concentrations. At the start of each depletion step, a required k (near unity) may be achieved by adjustment of control rod poison, as shown in Fig. E.2. The end of a cycle is established by extrapolation to zero poison concentration at a specified final reactivity.

At the end of a cycle, material in one zone is discharged and reprocessed, and when desired, fuel nuclides are returned to the core at some later cycle. If the end of a cycle occurs at a time that exceeds a specified total accumulated time, the reactor history is ended. The fuel-cycle cost is then calculated from a record that has been kept for each cycle. For each nuclide of interest, this record has tabulated the inventory at the start of the cycle, the inventory at the end, and the removal at the end of the cycle, with consideration given to recycle. The cost calculation utilizes the discounted-worth method based on private ownership of fuel.

Individual core concepts require individual attention. A general approach was incorporated in TONG to solve the eigenvalue problem by outer iteration on nuclide concentrations; such calculation was necessary to adjust the moderator ratio of the SSCR. For poison or fuel search calculations, a much faster direct iterative procedure was used that treats the nuclide concentration (rather than the multiplication factor) as the eigenvalue of the problem. The broad-group absorption cross sections of a single nuclide may be correlated as a function of the moderator composition. This correlation was used in the SSCR calculations to obtain the thorium resonance cross sections at each moderator ratio.

For the HWR problem the code permits depletion of nuclides in more than one location in the cell by treating them as separate nuclides with different broad-group cross sections. Thus, effects of shielding of the fertile material resonances in the inside rings of a multiring fuel element may be examined.

For the HTGR problem the code allows recycle of bred material once only and then discharge (sale). Also the initial loading of fuel may be recycled or not as desired.

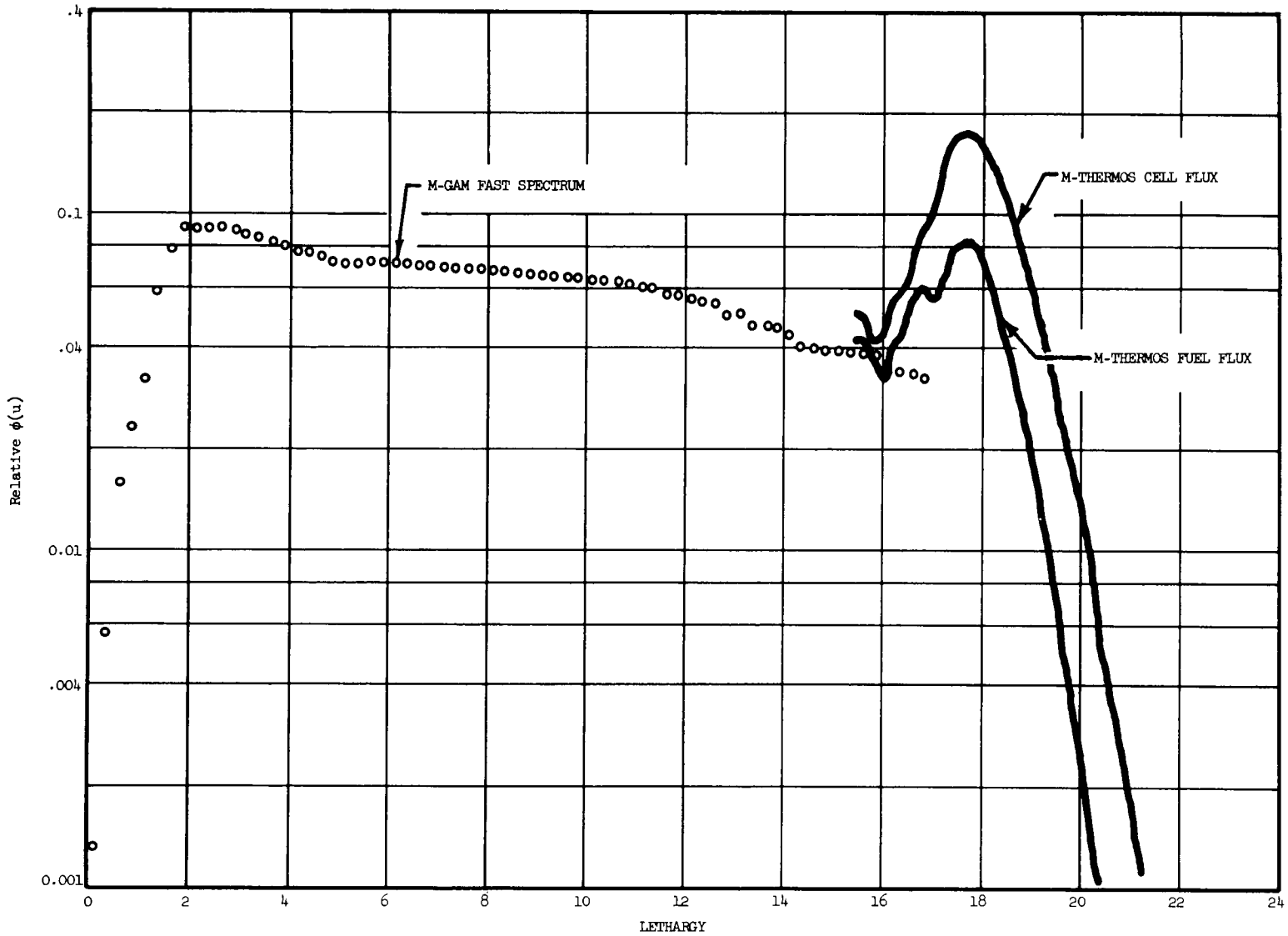

Fig. E.1. Lethargy Flux Spectrum for SGR as Computed by TONG.

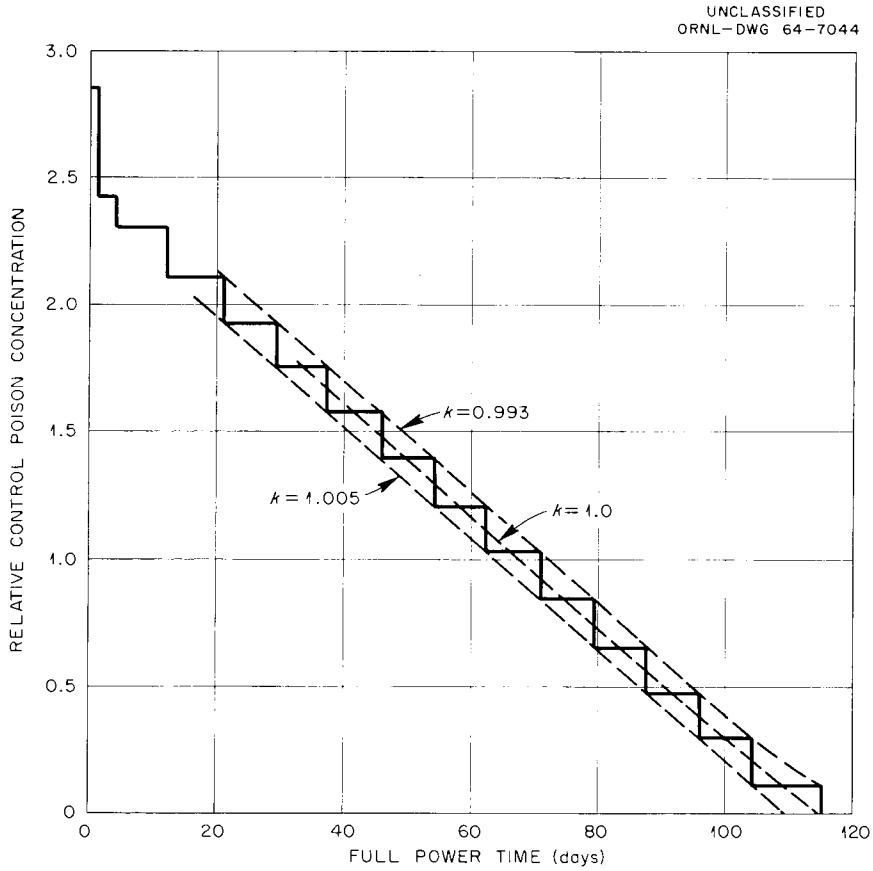

Fig. E.2. Example Cycle History Calculated by Tong.

In one HTGR history calculation that considered 12 zones, over 1000 nuclide concentrations were determined at each of 580 points in time; an eigenvalue problem was solved at each by adjusting nuclide concentrations to satisfy a desired k. About 3 hr of IBM-7090 time was used for an individual history. Machine time required to establish a 30-year history for a nonrecycle core that attained equilibrium in a few cycles (consider about 12 cycles for complete equilibrium with three-zone partial refueling) was about 20 min to prepare microscopic cross sections and 5 to 10 min for each history generated, including calculation of fuel-cycle cost.

A particularly useful feature of this code is that nuclide identifications are not built into the code but, rather, are specified by input identification numbers. It is thus possible to consider independently nuclides that are actually the same; for example, one U^{235} nuclide may be recycled material, while another is fresh feed. This flexibility was achieved by using a general treatment of the chain equations that requires all information about each chain to be specified by data supplied. Coupled chains and partial capture routes are also allowed.

Computer Code Assault

The IBM-7090 code ASSAULT was developed at ORNL to treat the two-dimensional multineutron-group depletion problem. This code uses the line-balance, overrelaxation scheme of the code EXTERMINATOR[1] and the depletion calculation of the code TONG. To treat the poisoned core problem, the concentration of the poison is considered to be the eigenvalue of the problem, and an iterative calculation proceeds directly toward the result to satisfy a desired multiplication factor.

Concentrations of nuclides are determined in regions containing several space mesh points, and exposure is to the average neutron flux in each region. To keep the calculation of nuclide concentrations within the fast memory, the total number of nuclide concentrations must be limited to about 12,000.

Machine time for such a calculation is quite dependent on any particular problem. Succeeding eigenvalue calculations require much fewer iterations than a first one, since a good initial estimate of the flux distribution is available. This code has an extrapolation routine that permits the final answer to be approached quickly after the point values begin to move in nearly a single mode. Data requirements are sufficiently complex (due to required specification of nuclide capture-and-decay chains and fission yields) that rerun of a problem is not unusual to correct the results for data errors; the flexibility and capability are considered to offset the disadvantages.

Microscopic cross sections used in ASSAULT calculations are prepared by the code TONG.

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[1]T. B. Fowler, M. Tobias, and D. R. Vondy, EXTERMINATOR — A Multigroup Code for Solving Neutron Diffusion Equations in One- and Two-Dimensions, USAEC Report ORNL-TM-842, Oak Ridge National Laboratory (in preparation).

Appendix F

BASIS AND CERTAIN FEATURES OF THE DISCOUNT TECHNIQUE

D. R. Vondy

If the past history of all outlays of money and revenue were available for a company, it would be simple to display an economic picture for the period. Given a set of ground rules and assumed conditions, a similar display can be made for the future history of a proposed operation. The discounted worth method of economic analysis eliminates the time dependence of each contributing cost or revenue from this type of analysis and thereby reduces it to a simple balance without need for trial-and-error calculation. This discussion presents a simplified derivation of the discount equations applicable to economic analysis of a private electric utility and explains certain methods of calculation employed in this study.

It is assumed that revenue from an investment over a certain period will retire all associated indebtedness, as well as cover all costs. In actual practice, the services of a utility company normally increase with time, so there is not really retirement of debt because new investments are made that bring about a total increase in debt. Thus, "retirement of debt" actually means "freeing the money for new investment," but this does not alter the calculation. It is assumed that the indebtedness is in a fixed ratio of stock to bonds and that interest on bonds is tax deductible, while return on stock is not; a fixed ratio of indebtedness is realistic when only a fraction of a company's operation is to be examined.

While careful consideration is given here to the payment of income tax, there are many complications, such as local taxes, that are avoided to preserve clarity. The less favorable sinking-fund method of depreciation and the more favorable sum-of-the-digits method add complexity; the more elementary straight-line method (fixed periodic depreciation) is used. It should not be interpreted that these factors which are avoided are not important or that they cannot be handled; they are omitted only for simplification. In the analysis, "operating costs" are those that are immediately tax deductible, whereas "investment" or "capitalized expenditures" are those that can be deducted only as they are depreciated.

Income and outlay are assumed to occur at the end of each accounting period. There will be an outstanding debt at the end of each period that is to be eliminated at the end of the history. The unit price of electricity that is required to retire this debt is taken as constant over the plant life.

The following list defines symbols used for an accounting period n:

$Q(n)$ = amount of energy sold during period,
$Y(n)$ = outstanding indebtedness before considering income and outlays during period,
$Z(n)$ = investment (capitalized expenditure),
$V(n)$ = income from other than energy sale,
$D(n)$ = depreciation,
$O(n)$ = deductible operating costs,
$T(n)$ = income taxes,

$R(n)$ = net retirement income after costs and taxes,
C = direct cost before interest,
I = interest charge, which includes real cost of indebtedness and taxes,
P = unit selling price of energy to return all investment costs,
X = discount factor defined by the development,
N = history life,
r = tax rate on taxable income,
i = required return on stock,
j = required return on bonds,
b = fractional indebtedness in bonds,
m = fixed charge or interest on an investment.

Income tax is given by the applicable fraction of taxable income:

$$T(n) = r\,[P\,Q(n) - D(n) - O(n) - jb\,Y(n)]\ . \qquad (F.1)$$

Net income is that remaining after costs:

$$\begin{aligned} R(n) &= P\,Q(n) + V(n) - O(n) - [jb + i(1-b)]\,Y(n) - T(n) \\ &= (1-r)\,P\,Q(n) + V(n) - (1-r)\,O(n) + r\,D(n) \\ &\qquad - [j(1-r)b + i(1-b)]\,Y(n)\ . \qquad (F.2) \end{aligned}$$

Reduction in the outstanding debt is achieved by applying net income:

$$\begin{aligned} Y(n+1) &= Y(n) + Z(n) - R(n)\ , \\ &= [1 + j(1-r)b + i(1-b)]\,Y(n) + Z(n) \\ &\qquad - (1-r)\,P\,Q(n) - V(n) + (1-r)\,O(n) - r\,D(n)\ . \qquad (F.3) \end{aligned}$$

Equation (F.3) is one of recurrence in the outstanding debt. Recognizing the terms other than Y to be independent of Y, it may be simplified to

$$Y(n+1) = (1+X)\,Y(n) + A(n)\ , \qquad (F.4)$$

where

$$X = j(1-r)b + i(1-b)\ . \qquad (F.5)$$

For an initial investment and indebtedness of $Y(1) = Z(0)$, the solution to Eq. (F.4) is given by the expression

$$Y(a) = \sum_{n=0}^{n=a-1} (1 + X)^{a-n-1} A(n) ,$$

and retiring all indebtedness, $Y(N + 1) = 0$, is given by the expression

$$\sum_{n=0}^{N} (1 + X)^{N-n} A(n) = 0 . \qquad (F.6)$$

In terms of the primary variables, the solution is given by

$$\sum_{n=0}^{N} (1 + X)^{N-n} [Z(n) - (1 - r) P Q(n) - V(n) + (1 - r) O(n) - r D(n)] = 0 . \qquad (F.7)$$

The solution of Eq. (F.7) for an unknown unit selling price of energy is

$$P = \frac{\sum_{n=0}^{N} (1 + X)^{N-n} \left[\frac{Z(n) - V(n)}{(1 - r)} + O(n) - \frac{r}{1 - r} D(n)\right]}{\sum_{n=1}^{N} (1 + X)^{N-n} Q(n)} . \qquad (F.8)$$

Equation (F.7) discounts all items to the end of the history, that is, future value discounting; it is generally more flexible to work with present value discounting, which is obtained by multiplying the numerator and denominator of Eq. (F.8) by $(1 + X)^{-N}$:

$$P = \frac{\sum_{n=0}^{N} (1 + X)^{-n} \left[\frac{Z(n) - V(n)}{(1 - r)} + O(n) - \frac{r}{1 - r} D(n)\right]}{\sum_{n=1}^{N} (1 + X)^{-n} Q(n)} ; \qquad (F.9)$$

at beginning of reactor life

$$D(0) = O(0) = Q(0) = 0 .$$

The denominator may be interpreted as the present amount of power, and if $Q(n)$ is independent of n, it is the present value of an annuity. The factor X is the discount factor given in this simplified analysis by required returns, tax rate, and indebtedness split; it may be interpreted as the interest charge on outstanding debt after taking into consideration that bond payments are deductible, as indicated in Eq. (F.5):

$$X = j(1 - r)b + i(1 - b) \ .$$

For example, if $r = 0.48$ and $b = 0.67$, $X = 0.35j + 0.33i$; for $j = 0.045$ and $i = 0.09$, $X = 0.045$. The required return on investment for a number of depreciation methods based on annual accounting is presented in Fig. F.1 as dependent on the discount factor X; certain contributions that add directly to the required return, such as local taxes, insurance, and replacement, are not included.

The discount factor given by Eq. (F.9) may be compared with the "cost of money" factor, $jb + i(1 - b)$. Thus the true discount factor is less than the cost of money by an amount rjb. This may be explained by the fact that payments on bond indebtedness are tax deductible and any increase in the ratio of bond to equity capital effectively lowers the discount factor. Use of a discount factor other than that given by Eq. (F.9), moreover, will not give the correct present value of future expenses and receipts.

In this study, cost items such as fuel purchases, fabrication, processing, and shipping are considered to make two contributions to selling price, direct cost plus interest. The direct cost component is the actual cost divided by the amount of power produced, and "interest" is the difference between the total cost from Eq. (F.9) and the direct cost:

$$P = C + I \ . \tag{F.10}$$

For an initial investment that is depreciated linearly over its life, such that $D(n) = Z(0)/N$, an assumed fixed energy delivery rate, $Q = Q(n)$, and using the subscript Z to denote treatment as a capitalized expense, Eq. (F.9) gives

$$P_Z = \frac{Z(0)}{(1 - r)NQ} \left[\frac{N}{\sum_{n=1}^{N} (1 + X)^{-n}} - r \right] ;$$

and, since $C_Z = Z(0)/NQ$,

$$I_Z = \frac{Z(0)}{(1 - r)NQ} \left[\frac{N}{\sum_{n=1}^{N} (1 + X)^{-n}} - 1 \right] . \tag{F.11}$$

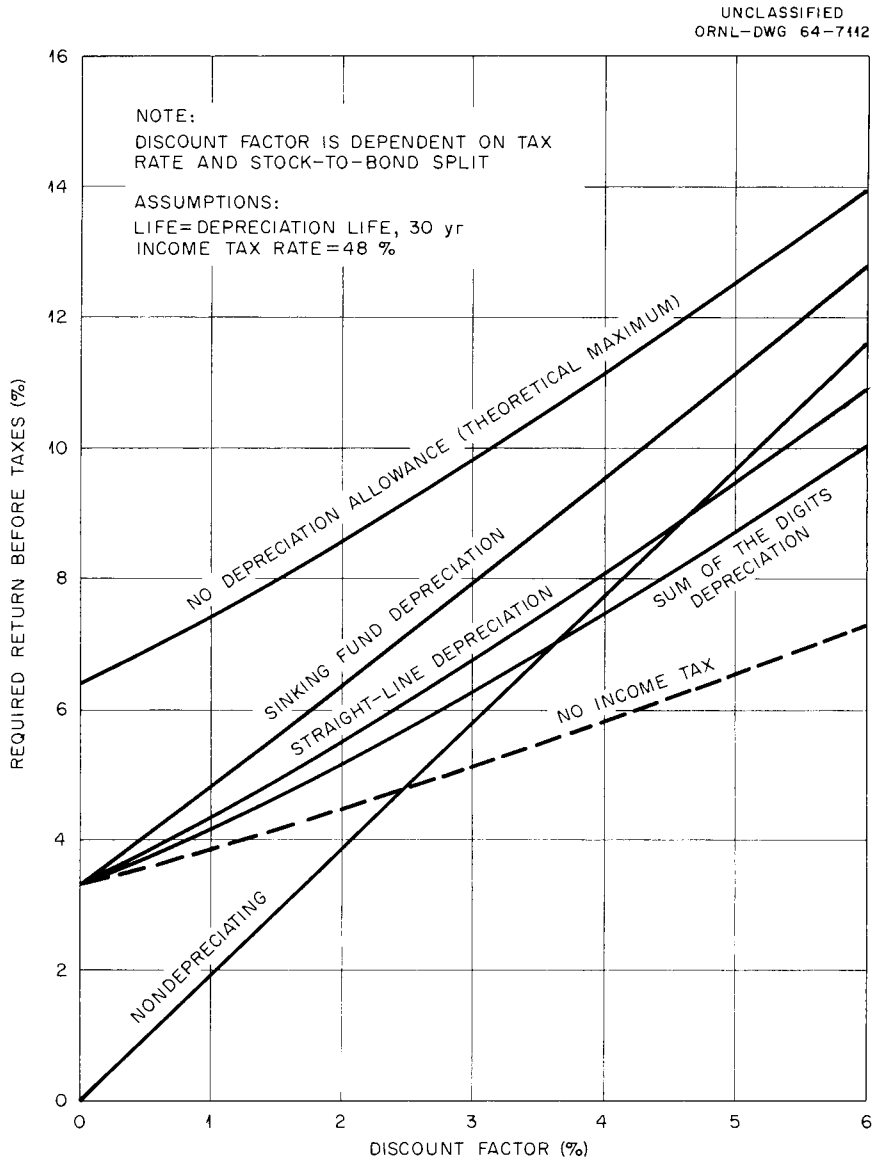

Fig. F.1. Return Required on Capital Investment.

If this investment had been treated as a deductible operating cost, O(O), rather than capitalized and depreciated, the interest would be given by difference. Using the subscript O to denote treatment as an operating expense, the interest charge is

$$I_O = P_O - C_O = \frac{O(O)}{NQ}\left[\frac{N}{\sum_{n=1}^{N}(1+X)^{-n}} - 1\right] . \qquad \text{(F.12)}$$

For $Z(0) = O(0)$, the ratio of the interest charge for linear depreciation to that for deduction immediately as an operating cost is shown by the ratio of Eq. (F.11) to Eq. (F.12) to be dependent only on the tax rate:

$$\frac{I_Z}{I_O} = \frac{1}{1 - r} \; . \tag{F.13}$$

If other factors besides cost of money and income tax are to be charged against outstanding debt, or if some arbitrary interest rate, m, is used as in this study, Eq. (F.14) applies:

$$\frac{I_Z}{I_O} = \frac{m}{X} \; . \tag{F.14}$$

The interest charge may be calculated as if costs were deductible operating costs, and this charge may be corrected by the ratio of the specified interest charge to the discount factor.

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