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REPORT

of the

FLUID FUEL REACTORS TASK FORCE

to the

Division of Reactor Development,

United States Atomic Energy Commission

February 1959

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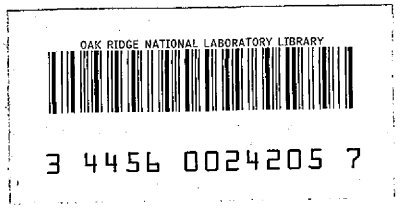
PREFACE

This report was prepared by a task force composed of fifteen highly qualified engineers and scientists from the Atomic Energy Commission, its national laboratory and contractor organizations, and representatives of the architect-engineering and utility industry. The task force was named the Fluid Fuel Reactors Task Force and was organized and convened by the Evaluation and Planning Branch of the Office of Civilian Reactors to perform a critical evaluation of the three fluid fuel reactor concepts (aqueous homogeneous - AHR, molten salt - MSR, and liquid metal fuel - LMFR) under development by the Commission.

The Task Force met continuously during January and February of 1959 and evaluated information presented by the national laboratories and industrial contractors developing the concepts. This document is the report of the Fluid Fuel Reactors Task Force to the Division of Reactor Development and represents the group evaluation and judgment of the three fluid fuel reactor concepts.



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I. INTRODUCTION

The Commission organized the Ad Hoc Advisory Committee on Reactor Policies and Programs in the Fall of 1958 to:

> "(1) review the Commission's civilian power program;
>
> "(2) advise the Commission in connection with the formulation of a sound basic policy in the light of our current economy and the present stage of development of nuclear power technology;
>
> "(3) assist the Commission in establishing new goals and redefining the problems connected with the foreseeable expansion of commercial utilization of nuclear power and the concomitant development of a vigorous nuclear equipment manufacturing industry; and
>
> "(4) recommend immediate and long-range programs to achieve the Commission's new goals."

The Committee presented their report to the Commission, January 2, 1959. One of their recommendations for the technical program in the near future is reproduced below:

> "The aqueous homogeneous, molten bismuth, and molten salt reactors all offer the possibility of reducing the cost of the fuel cycle, and the last two offer the possibility of high temperature operation. These three concepts for power reactors should be critically compared and work concentrated on the concept that appears the most promising."

Further comments of the Committee dealing with Fluid Fuel Reactors are reproduced in Appendix A.

The Evaluation and Planning Branch of the Civilian Reactors Office, Division of Reactor Development, organized and convened the Fluid Fuel Reactors Task Force in January, 1959, to make the comparison of the aqueous homogeneous, molten salt, and liquid metal fueled reactor concepts in accordance with the Ad Hoc Committee's recommendations.

The Task Force consisted of the following members:

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The Atomic Energy Commission wishes to take this opportunity to thank the members for their devoted and diligent participation on the Task Force.

II. OBJECTIVES OF FLUID FUEL REACTORS TASK FORCE STUDY

The Evaluation and Planning Branch of the Office of Civilian Reactors, Division of Reactor Development requested the Task Force to make a relative comparison of the fluid fuel reactor concepts (aqueous homogeneous--AHR, molten salt--MSR, and liquid metal fuel--LMFR) to determine the:

a. present state of development and technical feasibility,

b. technical feasibility of breeding,

c. potential power cost in mills per kilowatt-hour in a system optimized for power production, and

d. research and development program (direction, timing, and cost) necessary to produce a reactor capable of achieving the cost potential mentioned above, for each concept.

The Evaluation and Planning Branch requested enough information of a comparative nature to enable the Atomic Energy Commission to choose the proper course of action and magnitude of support that would be required to develop each type.

III. SUMMARY

A. Present State of Development and Technical Feasibility

The molten salt reactor has the highest probability of achieving technical feasibility. This is largely due to the use of a solution fuel (as contrasted to a slurry fuel in the LMFR and the AHR), and the availability of a suitable container material (INOR-8).

Summaries of the relative comparisons of the three concepts follow:

1. The technical feasibility of fuels and materials is a critical factor. At the present state of technology, the MSR has the best possibility of obtaining a satisfactory fuel, if indeed it does not already have a satisfactory fuel. Slurries, as used by the LMFR and AHR require a greater amount of development effort to establish feasibility. The MSR also offers the best possibility for achieving a satisfactory container material since the LMFR and the AHR have difficult materials problems at the present stage of technology. However, the compatibility of molten salt fuel with graphite which is contemplated for use for the internal construction of a reactor still remains to be demonstrated and the problem is judged to be more severe than in the LMFR.

2. The achievement of satisfactory primary and auxiliary systems and components depends largely on matters of engineering ingenuity and is believed to be technically feasible; however, these systems will be complicated and hence expensive. In comparing the molten salt with the liquid metal fuel reactor, no significance has been placed on difficulties arising from the molten salt solution's higher melting point (975°F vs. 525°F) and higher top operating temperature (1225°F vs. 1050°F). Difficulties in design caused by the higher temperatures are offset by the fact that the MSR primary system will be smaller than the LMFR system because of the higher volumetric heat capacity of the salt.

3. From the standpoint of operation, it is anticipated that all three reactor concepts can be designed to meet load changes, and it is assumed that they will be able to operate for extended periods of time. However, this has not yet been demonstrated. When considered from the standpoint of reliability for extended periods of time, the AHR is at a disadvantage because of its more extensive and complex auxiliary systems.

4. Maintenance is the most important factor influencing the practicability of any of the three concepts. At present the feasibility of maintaining a large fluid fuel reactor power station is doubtful because of the need for circulating a high-level radioactive, fluid fuel stream. While experience with the HRE-2 has shown that it can be maintained by the use of wet maintenance, it is not known if these techniques can be applied to large plants. The use of remote dry maintenance for any of the plants is unproven. The feasibility of any maintenance scheme can only be established by a comprehensive design study backed up by extensive full-scale mockup testing under conditions simulating actual requirements.

5. The AHR is easily controlled, as has been indicated by reactor experience. Analytical studies show that the MSR and LMFR can be controlled but this requires experimental verification. Both may require shim rods.

6. The AHR is potentially the most hazardous because of its high pressure system, radiolytic gas explosion hazard, and the potential instability of the fuel. However, HRE-2 has operated for a long period of time under very disadvantageous circumstances without serious release of radioactivity. The MSR and LMFR are similar to each other in their safety characteristics.

7. With the exception of the AHR, chemical reprocessing for the reference designs has received little attention.

B. Technical Feasibility of Breeding

The evaluation of breeding in the fluid fuel systems on the thorium-U^{233} cycle leads to the following conclusions:

1. One-region reactors can breed only in large sizes.

2. It is highly probable that all three of the systems can be developed into "hold own" breeders. Likely breeding ratios for the AHR, LMFR, and MSR are 1.09, 1.05, and 1.05, respectively.

3. On the basis of a reasonable extrapolation of the current development program, only the AHR has the possibility of achieving a reasonably short doubling time (of the order of 15 years).

4. The LMFR and MSR can also achieve reasonable doubling times if internally cooled. However, the required development programs make the internally-cooled, non-aqueous doublers considerably farther away in terms of both time and money than is the AHR doubler.

5. The uncertainty in eta for U-233 does not seriously jeopardize the probability of all three systems being "hold own" breeders, but it is crucial in the feasibility of the doublers.

C. Power Costs

The cost tabulations summarized below and presented more fully in Section VII were prepared to provide a basis for evaluating the relative power cost potentialities and indicated level of investments of the molten salt, liquid metal fuel, and aqueous homogeneous reactor concepts for large central power stations.

Summary of Power Costs
(For Reference Reactors: 333 MWE, gross)

(Based upon Table VII-1: Overall Summary of Power Costs)

	Mills/Kwh		
	LMFR	MSR	AHR[f]
Power Plant Investment [a]	5.72	6.24	6.70
Chemical & Waste Disposal Plant [b] Investment	.71	.71	.89
Fuel Inventory Use and Burnup [c]	1.36	1.37	.90
Chemical Plant Operation & Maint. [d]	.76	.76	.96
Power Plant Operation & Maint. [e]	1.46	1.57	1.66
Total Power Costs - Gross	10.0	10.7	11.1
Total Power Costs - Net	10.7	11.1	11.5

Notes:

a/ For details and explanation of power plant investment data, see Section VII and Table VII-2.

b/ For details and explanation of chemical processing and waste disposal plant investment figures, see Section V. For divergent views of Project Directors on costs of chemical processing, see pages 76, 95, and 169.

c/ For details and explanation of fuel costs, see Section VII, and Table VII-3.

Notes: (cont'd.)

d/ The chemical processing plant operation and maintenance costs are computed as 15% of the chemical processing plant investment shown above, based on Section V, part G. For divergent views of Project Directors on costs of chemical processing, see pages 76, 95, and 169.

e/ Operating costs are based on estimates of personnel requirements and other expenses. Power plant maintenance has been estimated at 3% of total power plant investment, including distributive items.

f/ Costs shown for AHR are for two-region solution core power breeder. Lower costs estimates for slurry fueled reactor are presented in Section XI, page 168.

The level of the cost data presented on the preceding page clearly is substantially higher than costs indicated in some earlier reports on fluid fuel reactor concepts.

In general, the data do not indicate any very substantial overall difference in the relative cost potential of the three concepts.

For valid comparison of costs for these reactor concepts with costs of other reactor concepts or with costs of conventional plants, it is of the utmost importance to take into account the differences in the many factors and conditions entering into the various estimates.

Conclusion

In determining fluid fuel reactor programs, the cost data presented herein should be regarded as secondary to technical feasibility and possibilities for advances by research and development.

The indicated costs are not so high as to preclude a reasonable research and development effort to develop and improve fluid fuel technology.

D. Research and Development Programs

The Task Force did not prepare a detailed comparison of the required research and development programs for the three reactor concepts. The programs presented were prepared by the individual Project Directors and all programs include an allowance for design, construction, and operation of a reactor experiment and a prototype to demonstrate feasibility as a commercial power plant. The programs presented each require in excess of one hundred million dollars and ten years of concentrated effort.

After reviewing the problems, the Task Force believes that any program of development on fluid fuel reactors should have as its objective the demonstration of technical feasibility.

The practicality of these concepts depends on the ability to construct these plants in a maintainable configuration at a competitive cost. Therefore, the programs should include extensive design and cost studies of a representative large-size plant, with particular emphasis on remote maintenance of primary and auxiliary system components. Simultaneously, the program should be directed at solving the research and development problems. Until these major uncertainties are resolved, however, no large-size prototypes should be constructed.

Any of the concepts that are pursued should follow the logical sequence of experiment and prototype before a commitment to construct a large-scale reactor is made.

IV. REFERENCE REACTOR SYSTEMS

The Project Directors were requested to furnish as complete information as possible on a reference reactor system design optimized for power production. Brief descriptions of these reference designs are presented below. More detailed information can be found in the project description sections - IX, X, and XI.

A. Molten Salt Reactor

The reference design molten salt reactor is an INOR-8 vessel containing a graphite assembly 12.25 feet in diameter by 12.25 feet high, through which molten salt fuel flows in vertical channels. The fuel salt is a solution composed of 0.3 mole percent UF_4, 13 mole percent ThF_4, 16 mole percent BeF_2, and 70.7 mole percent Li^7F. The fuel salt is heated from 1075°F to 1225°F in the core and is circulated from the reactor vessel to four primary heat exchangers by four fuel pumps. A barren coolant salt is used as the intermediate heat exchange fluid, which superheats and reheats steam in a Loeffler boiler system. The reactor develops 760 MW of heat. By using 2000 psig, 1000/1000°F steam in a reheat cycle, the net electrical output is 318 MWE.

B. Liquid Metal Fuel Reactor

The reference design liquid metal fuel reactor is fueled with a slurry containing approximately 3 w/o ThO_2-UO_2 in bismuth. The reactor core vessel is 14 feet in diameter and 14 feet high, containing a graphite core and reflector assembly. The fuel slurry is heated from 750°F to 1050°F through channels in the reactor core and circulated through three primary heat exchangers by three primary, variable speed pumps. The reactor develops 825 MW of heat. Sodium is used as the intermediate heat transfer coolant. The steam from the secondary heat exchangers supplies a non-reheat, 2000 psi, 1000°F steam turbine which produces 312 MW of net electricity.

C. Aqueous Homogeneous Reactor

Three concepts of aqueous homogeneous reactors were presented to the Task Force: a two-region, solution core, slurry blanket reactor; a two-region, slurry core, slurry blanket reactor; and a one-region slurry reactor. The fuel carrier and moderator is

heavy water in all cases. The Project Director chose the two-region, solution core, slurry blanket as the reference design reactor. This reactor consists of a 4 ft. diameter by 12 ft. long zirconium alloy core tank in an 8 ft. diameter by 16 ft. long pressure vessel. A five g/l solution of uranyl sulphate in heavy water is circulated through the core of the reactor and a thoria slurry containing 1000 g Th/l is circulated through the blanket. Heat is removed from the core solution in two circulating systems, each of which contains a steam generator and a circulating pump. The blanket slurry is recirculated through one similar heat removal circuit. Steam is generated at 400 psia and 435°F and sent directly to the turbogenerator. The heat generation rate for one reactor is 380 MWT: 320 MWT in the core and 60 MWT in the blanket. Three reactors provide steam for one turbogenerator in the reference station to produce 317 MWE net. The slurry-fueled reactors are similar but larger, so for each type one 1140 MWT reactor satisfies the requirements of the reference station. Further details of the other systems are included in the project description, Section XI.

V. COMPARATIVE ANALYSES OF PRESENT STATE OF DEVELOPMENT AND TECHNICAL FEASIBILITY

The following section presents an appraisal of the problems which must be solved before a successful reactor can be built and operated. It should be noted that considerably greater research and development effort has been applied to the aqueous homogeneous power reactor concept than to the other two. As a consequence, it is likely that a greater proportion of its problems have been uncovered than for the others. Since these problems have been noted and weighed in the following analyses, it is possible that this concept has been judged more severely than the others. No alternative treatment has been deemed possible.

A. Fuels and Materials

For the reference reactor designs presented for evaluation to the Task Force, two-region solution slurry aqueous homogeneous, one-region solution molten salt, and one-region slurry liquid metal - the molten salt system is judged to have fewest fuel and material problems with respect to both the current state of technology and the development effort required before technical feasibility can be established. The following table lists estimated relative technical difficulties for the three systems. Equal weight is given to fuel and materials, with the latter broken down to materials used for reactor internals, and all other materials. The larger the number rating, the smaller is the estimated probability of technical success.

Relative Technical Difficulties

	AHR		MSR	LMFR
	Solution Core (Reference)	Slurry Core (Alternate)		
Materials				
Internals (Graphite & Zircaloy)	5	2	2	1
Externals (SS, INOR-8, Croloy	3	2	1	3
Total Materials	8	4	3	4
Fuel	7	5	2	4
Summations	15	9	5	8

Discussion:

1. Materials

a. Internals (Graphite for MSR & LMFR - Zircaloy for AHR)

The graphite internals of the LMFR have fewer problems than those of the MSR system. The reason for this is that the fuel in a solution will have more tendency to penetrate the graphite than will the fuel in a slurry. It must be demonstrated whether such penetration can be avoided and if not whether adverse effects on irradiation will result. These data can be obtained only in long-time, in-pile loop tests. Accelerated tests may give useful information but the effect of time under irradiation may be equally or even more important than integrated flux.

The ability of a graphite structure 12-14 feet in diameter by 12-14 feet high to maintain its structural integrity for 30 years under conditions of high flux, temperature gradients, hot spots resulting from fuel collection in "dead spots," etc. can only be guessed. An important aspect of this problem is the degree of engineering ingenuity that is put into the design.

Despite these problems, the graphite in LMFR is assigned a low relative difficulty, 1. Perhaps the graphite in the MSR is twice as difficult a problem, due to more probable penetration by fuel (mentioned above), and the requirement to avoid serious reaction between oxygen and water (absorbed in the graphite) and the fuel.

The Zircaloy core tank for the two-region solution core homogeneous reactor is considered much more of a problem than the graphite in either the MSR or LMFR. Under conditions of the most recent design proposal, -0.025mUO_2SO_4 solution, 50 KW/l power density in solution near wall, and with a core tank wall temperature of 260°C (obtained by external cooling with the blanket slurry plus careful hydrodynamic design inside) - the average corrosion rate expected under normal operating conditions and at a velocity of 15 ft/sec is approximately 15 mils/year. This rate will increase to about 17 mils/year at 5 ft/sec flow rate. When, for design purposes, a safety factor of two is applied to these rates, the life of the core tank is about 15½ years on the basis that it is tolerable to corrode completely through a ½-inch wall.

Much more rapid penetration of this core vessel will occur if control of the conditions is not good enough to avoid inadvertant local heating. If high temperature develops and causes deposition of uranium rich solids or formation of the uranium rich second liquid phase, further rapid increase in local temperature will occur. Hence, the relatively poor rating of 5.

b. Externals

INOR-8 (79% Ni, 17% Mo, 7% Cr, 5% Fe), the basic container and piping material for the molten salt system, appears to offer basically a more trouble-free system than either the LMFR or AHR. Its corrosion and mass transfer properties are excellent; no measurable attack (less than ½ mil) occurring after one year exposure to salt at the maximum operating temperature. Weldability and fabricability are comparable to Inconel. The best rating, 1, was assigned to the MSR.

Although a less expensive material is proposed for the LMFR, 2¼ Croloy, there is a more definite problem with mass transfer of metal from hot to cold regions of the reactor system. Additives have minimized but not eliminated the problem. It has not yet been shown whether this Croloy will be a satisfactory material of construction for a delta T of 167°C. If Croloy requires a lower delta T for satisfactory operation, other costs will obviously go up. If a high strength steel, clad with carbon steel is required, it cannot be said the system can be constructed - if it is possible, the cost will be high. The lower rating of 3 is therefore assigned.

In the AHR, stress corrosion cracking is a worry in the presence of oxygen. In the presence of hydrogen, the stainless steel is not as corrosion-resistant as in oxygen-bearing slurries. There is also an increased possibility of hydrogen embrittlement for special purpose materials. For the uranyl sulphate solution, the stainless steel is dependent on relatively thick corrosion films. Although corrosion rates are very low (after initial corrosion), the films must be maintained to avoid rapid attack. In addition, 347 stainless steel is notch-sensitive and poses a design problem. The rating given is 3.

2. Fuels

The fuel for the molten salt system is considered superior to the AHR and LMFR fuels, basically because it is a solution. Although it shows promise of being satisfactory for long cycle times before requiring reprocessing there are some uncertainties that must be resolved, such as effect of oxygen on precipitation of uranium and the effect of precipitation of the noble fission product metals, Ru, Mo, and Nb. A good rating is assigned.

Both the AHR and LMFR have slurry problems which can hardly be defined at this time. Slurry technology is in its infancy and very considerable development effort is required before adequate information is obtained. It is doubtful whether slurries can have cycle times of the order of 10 years. In the LMFR the slurry particles

that are added to the system consist of 30-50% of UO_2, the balance ThO_2. For a long cycle time up to 50% of these metal atoms will fission, destroying completely the particle as such. It cannot even be guessed what this degradation will mean. The rating for the LMFR is 4.

In the AHR, a serious problem is also encountered with the uranyl sulphate solution. Extreme care must always be exercised to keep the solution within proper chemical control. Even a short time instability in the solution may result in burning a hole in the Zircaloy core tank with the consequence of mixing the solution and slurry fuels. This serious problem influences the rating number of both the core vessel and the fuel.

3. Alternate AHR

Also included in the table is a rating for the non-reference design, in which the core fuel is a slurry. The corrosion problem for Zircaloy seems to be substantially reduced in changing to this concept although data are quite meager. The better rating of 2 is assigned. Similarly, the corrosion resistance of stainless steel is improved, leading to a rating of 2, slightly better than that given for 2¼ Croloy in LMFR.

Elimination of the serious difficulties of the solution fuel makes possible a rating of 5 for the fuel, still somewhat worse than that for the LMFR because of the faster slurry settling rate.

Considering only fuel and materials, the slurry core has significantly less difficult technical problems than the solution core.

B. Primary System and Components

1. Reactor

The differences in the reactor concepts are dictated by the physical and nuclear properties of the fluid fuels. The AHR uses heavy water as the carrier fluid because of its good nuclear properties but the system must be operated at relatively high pressure to maximize the thermal efficiency. The size of the vessel is limited by this high pressure requirement and, for large power stations, multiple reactor units are used. For a single-region reactor, this size limitation also introduces thermal neutron shields. The LMFR and MSR utilize a low vapor pressure carrier but require the addition of a moderator.

The HRE-2 experience has shown that the hydrodynamic and heat transfer conditions at the surface of the container in contact with

uranyl sulphate solution must be carefully controlled to prohibit phase separation and eventually burnout of the inner container. In the reference concept the temperature of the inner container is controlled by removing heat with the slurry fluid flowing on the outside and with the inlet solution fluid flowing along the inside of the wall. The eventuality of credible accidents (e.g. loss of slurry flow) dictates that the solution fuel be capable of removing a large part of the heat generated in the inner wall; this requirement probably decreases the allowable temperature of the solution fuel and may further decrease an already unattractive thermal efficiency.

A considerable development effort has already been expended on the problems associated with the hydrodynamic and heat transfer conditions of solution fuels in spherical configurations. It has been shown that an acceptable design for a spherical container to accommodate solution fuel can be developed for steady-state operation. Higher total power at given maximum power density is attainable in the cylindrical configuration suggested for the two-region, solution-slurry reactor. Very little development work has been done on cylindrical configurations utilizing solution fuels and much more remains to be done.

In addition to the problems associated with the transient operations, it is highly probable that the inner container made of Zircaloy-2 will have to be replaced 10 to 15 years after startup as a result of the high corrosion rate by the solution fuel. This likelihood makes the use of the reference design reactor questionable for central power station installations.

The specification of a single-region or two-region reactor utilizing slurry fluids only, cannot be made at this time. Satisfactory materials for the long-term containment of slurry fluids in a reactor vessel, assuming proper design, are now available. However, the hydrodynamic and heat transfer characteristics of slurry fluids in several configurations are needed before the reactor vessel can be specified.

The LMFR and MSR concepts have problems which are similar to each other. The foremost of these is the requirement that the vessel and its internals be heated to a temperature in excess of the melting point of the fuel carrier in an inert atmosphere. The severity of the preconditioning problem is greater in the MSR because of the higher temperature and greater need to minimize oxygen contact with the fuel fluid. However, this difference appears to be insignificant. Careful design of the heating system is dictated to insure sufficient heating and acceptable thermal stresses in the reactor vessel and its internals.

The moderator chosen for the LMFR and MSR is high density, impervious graphite. It is necessary to provide support for the graphite cylinder and means by which the blocks are held firmly; these problems may involve considerable design effort and an acceptable design is expected to evolve from reactor experiment experience. Fuel and fission products may be deposited on and absorbed by the graphite in both the LMFR and MSR. The magnitude of this problem has been roughly defined for both reactors but the results obtained to date do not permit firm conclusions. The extent of the phenomenon and its possible consequences need to be examined experimentally and analytically. Several in-pile loops are indicated; the cost of these investigations is expected to be of the order of two million dollars.

The physical properties and heat transfer characteristics of the solution fuel suggested for the MSR have been determined experimentally. However, it appears prudent to determine the hydrodynamics of the particular reactor design in mockup studies. This work need not be extensive and can be accomplished for a modest expenditure.

Hydrodynamic investigations are required for the bismuth-thoria-uranium oxide slurry suggested for the LMFR. Extensive design work for the LMFR reactor is indicated because of the unknown properties of the slurry. It is conceivable that slurry hold-up in localized areas within the reactor may cause overheating and spalling of the graphite.

It appears that both the LMFR and MSR may require control rods for successful operation. The AHR does not require them. The need for control rods is a disadvantage but not a serious one.

It is apparent that additional development work must be performed for all three concepts. Since the fuel fluid for the MSR is a solution, it appears that design of a vessel for MSR will be most easily accomplished.

2. Primary System

The relative merits of the primary systems can be evaluated on the basis of the properties of the fuel carriers. From this standpoint the MSR fluoride salts have the advantage of high volumetric thermal capacity combined with a relatively low density and low vapor pressure.

The LMFR uses bismuth as the basic fuel carrier. The vapor pressure is low; however, the density is approximately 2½ that of the fluoride salts and the volumetric heat capacity is approximately 1/3 that of the fluoride salts.

Because of the high density of the fuel carrier and the low thermal capacity, the LMFR design is highly vulnerable to changes in allowable temperature difference in the loops. The result is a design based on a high difference. The consequence of decreasing the temperature difference is to increase the pumping horsepower requirement. This increase rapidly pushes the pumps beyond the present and anticipated future available canned motor capacities and would require additional loops.

The MSR on the other hand uses the high thermal capacity to obtain a low temperature rise across the reactor and hence a high mean temperature difference in the primary heat exchangers. There still remains flexibility in the design parameters within the limits of practicality should it prove necessary to drop the mean temperature difference in the exchangers and raise the reactor temperature difference.

The solutions or slurries used for the aqueous concepts have roughly the same volumetric thermal capacity as the fluoride salts but suffer from the high vapor pressures of water at the temperatures required for power production. In addition, the use of slurries in the two-region breeder concepts involves a difficult technology and presents a serious barrier to the development of components, particularly fuel circulating pumps.

The need for smaller piping in the case of the MSR tends to compensate for the higher temperatures in the thermal stress analysis. There is little to choose between the two concepts on this basis although the LMFR designers have apparently given more thought to this particular phase of the concept. This smaller piping also offsets the higher melting temperature of the molten salt. Thus the overall system preheating requirements are about equal for the two high temperature concepts.

The major components of all three concepts are pushing the frontiers of the technologies involved. The failure to solve any of the specific problems therefore jeopardizes the technical feasibility of the program. To date none of the concepts have demonstrated technical feasibility of utility-station-sized primary system components, however, the size problem can be minimized by using additional loops at some sacrifice of economic potential. It is believed that, in general, the technical feasibility of the components of the primary systems of the three concepts depends on matters of engineering ingenuity and are therefore evaluated as essentially equal.

3. Primary Auxiliary System

The success of any of these concepts necessitates the performance of functions by equipment not directly a part of the primary

heat transfer loop. Some of this hardware is in direct contact with highly radioactive fluids; other parts of it may become radioactive under some conceivable circumstances. All of this equipment must be considered to be highly contaminated, and will require remote maintenance.

Both the LMFR and MSR concepts suffer a serious disadvantage as a result of the relatively high melting point of the carrier materials. All of the primary system must be heated by an external source. The magnitude of the external source is of the order of 3 MW for both systems and will be about the same for both since the LMFR and MSR require temperatures of about 650°F and 1050°F, respectively, but the LMFR volume is about 1.7 times that of the MSR. Helium gas circulating in annuli about all primary equipment is used as a heating medium for both reactor systems; some other gas may be acceptable and an effort to eliminate the use of helium should be made because of its high price ($44/1000 cf at STP) and limited supply. The quantity of helium circulated is of the order of 3000 pounds per minute and moderate sized blowers at about 75 psig pressure are the prime movers. The blowers, furnaces, and piping must be carefully designed and constructed to eliminate wide temperature gradients in the primary equipment and leakage of the helium. It is likely that the circulating gas will become contaminated during operation. Therefore, provision for the remote maintenance of the blowers, furnaces and piping, and valves must be made; this provision complicates the design considerably. Design of this heating system is not impossible but it is expected that an acceptable design will only evolve from the design and construction of the actual equipment.

Either a corrosive solution or an erosive slurry--thoria in heavy water--or both flow through the primary auxiliary equipment of the AHR. Equipment for handling uranyl sulfate solution has been found. Some of the equipment must be titanium-lined to withstand the corrosive solution; this equipment has not been fabricated commercially in large sizes but such practices can probably be developed. The experience with the slurries has shown that it will be difficult to obtain the operating life of some of the equipment (particularly check valves) which is acceptable for use in a central power station. Difficult problems have been encountered and the results from some development work have not been encouraging. Additional development work is required and the indications are that the susceptible equipment will have to be eliminated insofar as is possible by design and by changing and complicating the mode of operation.

The primary auxiliary systems of the LMFR and MSR are characterized by large, but relatively simple, equipment whereas the AHR system involves several items of relatively small size and of specific function. Remote maintenance procedures for the LMFR and MSR have

not been demonstrated adequately and further development work is required. The problems of remotely maintaining the primary auxiliary systems for the LMFR and MSR are essentially the same as those involved with the primary heat transfer loop. The AHR primary auxiliary equipment can be maintained, as has been demonstrated by HRT experience.

The advantages and disadvantages are about equal for the three concepts. The distinguishing feature is the ease or difficulty with which the different problems may be solved. In this respect it is expected that the AHR slurry will use a low pressure system which is operated batchwise in an effort to minimize the problems caused by erosion. It is expected that the development of remote maintenance techniques will be successful but also difficult. It appears that there is little to distinguish the different concepts with regard to the primary auxiliary systems.

4. Instrumentation

The nuclear instrumentation associated directly with reactor operation is essentially the same for all three concepts. The state of this technology is highly developed and additional work directed to aid the operation of these three reactors does not appear to be required.

The primary loop and primary auxiliary loop instruments for the MSR and LMFR differ from those needed for the AHR. Measurement of a number of temperatures and several liquid levels appear requisite in the MSR and LMFR while in the AHR fewer of these measurements and flow rate and pressure measurements are needed. All of these instruments are exposed to intense neutron and gamma fields but this is not an insurmountable problem. All instruments must be designed to permit remote replacement; the instrument design for the MSR and LMFR is more difficult since the outer containment, heating and cooling shell must be penetrated.

Jacketing of the primary loop and some primary auxiliary equipment complicates the leak detection system, and instrumentation for this purpose is deemed necessary for the MSR and LMFR. The instrumentation to be used for this purpose has not been designed, but there are several possibilities. The need for this equipment is a definite disadvantage.

Some instrument development for both the MSR and LMFR is necessary and the extent of this development will become evident during design and operation of a reactor experiment. Development work has been needed for the HRT and it has been quite successful. Similar experience with the other reactor concepts can be expected.

Instrumentation requirements of the AHR are most easily met but mainly as a result of a successful development effort in connection with the HRE-2. The instrumentation requirements for the MSR and LMFR are nearly the same.

C. Operation

The primary purpose of any power producing scheme is the output of electrical power at the generator terminals. In attaining this objective, reliability and ease of operation of the plant are two of the most important factors. Both reliability and ease of operation depend on the design of the plant. Because a given plant is only a part of a much larger system, its operation will be affected by the operation of the entire system.

From the point of view of operating personnel, the people who will live with the plant after it is constructed, the rapidity and simplicity of start-up and shut-down operations is very important. Equally important is the ease with which the plant will respond to changes in electrical load. During the early life of the plant the necessity of shut-downs and start-ups will be dictated largely by the needs of maintenance. During low system demands, at night, and weekends the plant would carry a somewhat reduced load. As the plant becomes older and more efficient plants are added to the system it will no longer be base loaded. At this point in the plant's life, perhaps as little as ten years, it may be necessary to shut down during low demand periods.

The molten salt and liquid metal systems have the same basic problems in starting up and shutting down, both need a preheating and cooling system. For both the molten salt and liquid metal system the time required in a start-up operation will be determined by the time required to preheat the systems. It may be found in future operations of these plants as part of a large electrical system that it costs more to shut down and start up than to operate at a very low load. This would be a disadvantage for electrical systems with very low periodic loads. A possible alternative would be to hold the reactor just critical so the decay heat is equal to the system heat losses. All that would be required to put the plant back on system would be to start up the turbo-generator. For the aqueous homogeneous reactor system, the shut-down and start-up procedures do not present any great problems. The reactor plant is made subcritical by diluting the fuel and the rate at which the temperature can be reduced is determined by the permissible cooling rate of the system components; likewise, the start-up time is determined by the permissible heating rate of the components.

In comparison of the three systems from the standpoint of shut-down for maintenance and start-up, the aqueous homogeneous is by far

the simplest; the degree of difficulty is the same for both the molten salt and liquid metal reactor concepts.

Two of the proposed concepts have had the advantage of conducting reactor experiments, the Aircraft Reactor Experiment (ARE) and the aqueous homogeneous reactors (HRE-1 and HRE-2), both using solution fuels. From the standpoint of operability the experiments were a success. The operation of the aqueous homogeneous reactor is complicated by the need of very extensive auxiliary and supporting systems. The reliability of reactor operation will therefore depend on the proper functioning of these systems. Control rods are not necessary on this type reactor due to the large negative temperature coefficient. A serious problem with the solution type aqueous homogeneous reactor has been the phenomenon of large power surges during operation. This is due to fuel coming out and going back into solution in the core.

The operation of a slurry type reactor has yet to be demonstrated.

The Aircraft Reactor Experiment demonstrated the feasibility of operation of a high temperature molten salt reactor. Although the reactor was equipped with control rods it was found the reactor would follow changes in the heat extraction system due to the negative temperature coefficient of reactivity. For any large-scale plant, control would be a combination of the negative temperature coefficient, movement of control rods, and variation of fluid flow.

The liquid metal fuel reactor is at a disadvantage by not having had a reactor experiment. However, the operation of the reactor would probably be very similar to that of the molten salt.

The three concepts should be compared on the basis of reliable on-system operation in which the plant is expected to meet load changes and to operate for extended periods of time. All three concepts could equally well meet changes in load on the turbogenerator. The differences, if any, between the three would be the rate at which the changes could be met. In the case of reliability of operation for extended periods, the aqueous homogeneous is at a disadvantage because of its extensive auxiliary systems which are more complex than either of the other two concepts. Minor operational problems in these auxiliary systems could lead to a shut down of the plant.

D. Maintenance

Maintenance was immediately recognized as one of the most important factors influencing the practicability of any of the three concepts. The combination of unknown service life of components,

serious consequences resulting from release of radioactivity due to a failure, and extreme difficulty of hot maintenance dictate the plant arrangement, containment systems, and component design.

The molten salt and liquid metal systems offer the same basic problems in regard to plant layout and maintenance, with the possibility that the liquid metal system may be somewhat more difficult to maintain due to problems created by slurries in the system. This could be balanced out if appreciable air were to leak into the molten salt system, causing the precipitation of UO_2 from the fuel.

The maintenance concepts proposed for the above two systems can be used interchangeably in that the same fundamental problems and conditions are prevalent in both systems. Major equipment components, i.e. pumps and heat exchangers, appear to have been arranged so that replacements can be made without undue difficulty; however, it is felt that problems of instrumentation for pressure, temperature control, and leak control of the entire system have not been adequately solved from the standpoint of maintenance of this part of the system.

In layouts evaluated for LMFR and MSR, equipment has been stacked on several levels, one above the other. This, in turn, leads to dependence upon the effective performance of remotely operated robot vehicles, manipulators, television, and other controls to accomplish maintenance in the lower levels of the plant. This concept requires considerable spacing of the equipment to make certain that the robot carts have complete access to all parts. If this concept does not work, the only apparent solutions would be to spread the plant out on a horizontal plane so that complete visibility and access could be achieved from overhead or to install heavy viewing windows at each level and manipulators to provide access to each segment of the equipment. In either case the capital cost of the plant would be increased considerably.

While it is expected that the integrity of the piping systems in these concepts will be such that the frequency of remote welding will be low, the life of the plant could well depend upon the ability to remotely weld and inspect pipe of large size. A remotely operated welding machine has been designed, built, and has undergone tests. Considering the relatively short time the machine has been under development, the progress has been encouraging. Methods for remote inspection of welded joints are under investigation but no satisfactory solution has been demonstrated. The liquid metal, molten salt, and aqueous homogeneous concepts are not feasible unless remote welding or substitute methods of closure are perfected.

On the positive side, the molten salt and the liquid metal systems have some advantage over the aqueous homogeneous system.

While at operating temperatures they have all the characteristics of the liquid systems. When the salt and metal are allowed to cool down and freeze, they take on the characteristics of a solid fuel in that most of the radioactive fission products will be frozen in the metal or salt. This will minimize the problem normally expected in aqueous systems, that fission products will gradually contaminate large areas of the plant including the areas above the cell blocks, the maintenance tools, the operating cells, and, ultimately require elaborate filtration systems to prevent radioactive particulate matter from being discharged from the stack to the surrounding environs.

The aqueous homogeneous system requires more equipment, due to the necessity of having three reactor systems to achieve the same power production as one reactor in the other two systems. It also has the necessity of doing more frequent reprocessing of the fuel, with associated on-plant reprocessing facilities.

The aqueous system has one simplifying feature in that pipe and vessel jacketing are not needed as in the other two concepts. This jacketing increases several-fold the magnitude of the maintenance job in the liquid metal and molten salt systems.

The aqueous homogeneous reactor concept has other basic requirements necessitating additional maintenance: gas recombiners, D_2O recovery, daily (or frequent) operation of reprocessing and waste systems, and the much higher operating pressure. Based on present knowledge from the two reactor experiments of the aqueous homogeneous concept, it has been demonstrated that maintenance is possible, albeit time consuming, while the molten salt and liquid metal systems have yet to be demonstrated and, in fact, depend upon the successful completion of extensive development programs.

In conclusion, based on today's knowledge and the experience of HRE-2, a properly designed aqueous homogeneous system can be maintained by the use of wet maintenance, although at high cost and considerable down time. The use of a dry maintenance scheme for the molten salt, liquid metal, and aqueous homogeneous systems depend upon the satisfactory development of remotely operated maintenance equipment. The degree of difficulty and cost of maintenance between the three systems will depend upon the plant layout and the maintenance scheme adopted.

E. Control

All three reactor concepts rely upon a gross negative temperature coefficient of reactivity as the primary means of controlling the reactor power. Through this inherent feature some load changes are rapidly and safely accommodated. The LMFR concept has included

variable speed primary pumps to minimize wide and relatively rapid changes in reactor temperatures. While conservation of design may have been the reason for including this feature, it appears that the need for variable speed control is more acute in the LMFR than in the MSR. The temperature rise across the reactor at normal operating conditions is 300°F in LMFR vs. 150°F in MSR and the magnitude of the thermal shocks that could be imposed upon the system are larger.

The temperature coefficient in the AHR is about 100 times greater than that in either the LMFR or MSR. Reactivity changes in the order of 2 percent Δ k/k per second can be controlled safely in AHR. Control or safety rods are not included in AHR but are indicated for LMFR and MSR.

For concepts utilizing slurry fuels it has been calculated that it is possible to operate (at constant temperature) with slurry concentration such that any variation in the concentration will cause subcriticality. These calculations have not included the effects of localized variations of slurry concentrations; these effects can be ascertained by reactor experiments only. There are several unanswered questions with regard to control of the MSR and they, too, will require investigation by operating a reactor experiment.

In the operation of the AHR, fuel fluids may be continuously removed and added to the primary heat exchange loops. This arrangement permits rapid variation of fuel concentration to accommodate wide changes in load and insures maximum thermal efficiency during peak load periods. The high temperature reactor concepts do not have as rapid flexibility in accommodating load changes and it may be desirable to use control rods for this purpose.

The aqueous homogeneous reactor experiments have demonstrated that reactors of this type can be controlled easily. The MSR and LMFR have not had this opportunity.

F. Hazards

The AHR is potentially the most hazardous of the three fluid fuel concepts. The primary reason for this is that it is a high pressure system. Additional contributing factors are the radiolytic gas explosion hazard, and the fact that its fuel stability is probably the most precarious of the three concepts.

The two non-aqueous systems are quite similar in their safety characteristics. Neither Po-210 build-up in the LMFR nor tritium production in the MSR are considered overly serious problems. There

is a possible complication resulting from the use of a slurry in the LMFR.

A possible design limitation from the point of view of hazards is operation at very low values of the effective delayed neutron fraction, beta eff. It remains to be demonstrated that satisfactory operation can be obtained with extremely low values of beta eff.

The homogeneous core, as opposed to the core with coolant passages, is more susceptible to nuclear instabilities arising from hydrodynamic flow irregularities.

The AHR, though it seems to have more potential hazards, has the advantage of being able to point to existing experience, which often is reassuring.

Discussion:

Fluid fuel reactors are characterized by the requirement to pump, sample, and otherwise handle quantities of highly radioactive liquids (or slurries). This means that standards of integrity must be quite high and that precautions to prevent leakage of the fuels should be taken to the maximum extent commensurable with the requirement to operate (from the points of view of physical operation and of keeping the costs tolerable).

The hazards analysis is complicated by the many possible variations within each of the fluid fuel concepts. The variations within each concept often differ considerably among themselves in some of the most important characteristics concerned with reactor safety. For example, they may differ in such respects as:

(a) Solution vs. slurry fuel.

(b) Continuous or frequent removal of fission products vs. long-term accumulation of fission products within the system.

(c) Control rods vs. no control rods.

(d) The magnitudes and time constants of the various components of the temperature coefficient of reactivity.

(e) The effective delayed neutron fraction.

As a consequence of these variations, it is often difficult to make generalizations applicable to each of the concepts as a whole.

1. Nuclear Stability

Many theoretical analyses of the stability of circulating fluid fuel systems have been made. The results have invariably been that the systems would not suffer seriously from this cause. In those cases in which the power at which the system would become unstable was determined theoretically, it was found to be many times the normal power.

The theoretical studies have to some extent been verified by experimental investigations in the HRE-1, HRE-2, and ARE. Even though these investigations have been essentially reassuring so far as inherent stability is concerned, they have had their disquieting features. HRE-1 had reactivity variations due to flow pattern variations in the size of central flow vortex. In HRE-2, there have been reactivity oscillations, due in all probability to insertion of precipitated fuel into the system (and hence, not a proper mechanism within the definition of inherent nuclear stability). The ARE showed good nuclear stability characteristics.

From the above there is little reason to believe that inherent instability will become a serious technical feasibility question. However, the theoretical studies have been deficient in that the external feedback mechanisms were often ignored or treated only superficially. The analyses which did consider external feedback showed no serious instability.

Of the three concepts, the AHR is probably the most susceptible to nuclear instability. This is a consequence of its large negative temperature coefficient which, due to circulation effects, has a delayed component which may be a contributing factor to instability. Also, hydrodynamic fluctuations are most likely in the AHR because the flow pattern encompasses the entire core. In the other systems the flow proceeds in channels through the core, minimizing the possibility of important fluctuations. Proper hydrodynamic flow design will be an important problem in the development of fluid fuel reactors.

The experimental and theoretical stability investigation in future reactor experiments should be a central part of the program. Experimentally, this should probably include oscillator programs since the theoretical analysis is then more amenable to quantitative conclusions. There should also be a more extensive theoretical program for realistically considering external feedback.

2. Nuclear Accidents

The main categories of nuclear accidents to which these systems are subject are: (a) fuel accidents, (b) fluid temperature accidents, (c) structural integrity failures, and (d) control system mismanagements.

a. Fuel Accidents

These accidents arise from abnormal quantities of fuel in the core. This condition could arise from fuel accumulation in the core; from sudden loss of such accumulation; from deposition in the external system, with subsequent return of this fuel to the core. Examples of possible difficulties of this type are:

AHR

(1) Formation of uranium-rich liquid phase by local overheating; (2) precipitation of uranium, containing mixed sulphate brought on by build-up of dissolved nickel (most probable in hottest region); (3) in inclusion of uranium in (or absorption on) the corrosion product on zirconium alloy core vessel, or similar association with external system stainless steel solid corrosion product; (4) deposition of uranium-bearing residue by local boiling; (5) settling of slurry in reactor; (6) settling or caking of slurry in external system.

MSR

(1) Formation of solid $3LiF \cdot ThF_4$ (including some uranium), a consequence of too low a temperature; (2) precipitation of mixed trifluorides, when the sum of the concentrations of rare earths, trivalent plutonium and uranium (produced by reaction of the fuel with the nickel-bearing container material) exceeds the solubility limit; (3) precipitation of UO_2, UO_3, or UO_2F_2 by the inleakage of oxygen or water; (4) association of fuel with deposited fission products such as molybdenum and ruthenium (this seems unlikely, judging from the limited information available); (5) association of fuel with INOR-8 corrosion products.

LMFR

(1) Precipitation of uranium from the solution or partial freezing of the bismuth with possible greater than normal uranium content in the solid -- for solution fuel, the increased uranium content is expected; for slurry fuel it can be true; (2) accumulation of mass of slurry particles (presumably by flotation) in the reactor; (3) accumulation of mass slurry particles in external system; (4) possible contribution of above items by enough oxygen inleakage to prevent wetting of the slurry by bismuth; (5) association of fuel with iron and chromium deposited on cold areas (mass transfer).

In addition to accidents arising from fuel inhomogeneity, there is also the possibility of a fuel accident arising from a concentration error. Such wrong concentrations could arise from mismanagement during start-up.

In order to minimize the consequence of fuel accidents, it is desirable to operate at fuel concentration such that either increasing or decreasing the concentration reduces reactivity. In principle, this is sometimes possible, but in many cases such operation would too seriously compromise the design from other considerations. In addition, changes in fuel composition during operation seem to preclude the possibility.

A difficulty analogous to the fuel accident in its effect is the accidental change of flow rate through the core. A reactivity excursion can result from the changed number of delayed neutrons emitted in the core.

A complicating feature and a possible contributory factor to the fuel accident is the difficulty associated with an accurate accounting of the fuel inventory in the primary system.

b. Fluid Temperature Accidents

Such accidents can arise from mismanagement or misoperation of the heat transfer system. Due to its large temperature coefficient, the AHR is most susceptible to the cold slug accident.

Thus, the large negative temperature coefficient in the AHR has both advantages and disadvantages. It makes possible the introduction of large amounts of reactivity through cooled fluid entering the core. On the other hand, the large negative reactivity coefficient can act to quickly compensate large reactivity insertions.

c. Structural Integrity Failures

Failure of a core tank could allow core material to enter the blanket and thus give rise to a dangerous rise in reactivity. Other examples of accidents due to structural failure are the breaking off of a graphite section in the core of the non-aqueous systems or the transfer of secondary coolant material into the primary fuel stream. The latter accident is essentially impossible in the AHR due to the pressure differential of the primary and secondary systems.

d. Control System Mismanagement

At least the MSR and LMFR may require control rods. The main reason for this is that it may be awkward in these systems to reduce reactivity. Although the control rods provide additional protection in the event that the negative temperature coefficient is not sufficient to prevent a dangerous excursion, there is now the associated amount of reactivity available in the system and thus a possible source of difficulty.

A contributing factor to the likelihood of nuclear accidents is the small value of beta eff, the effective delayed neutron fraction.

Since the excursions that cause the most serious damage have reactivities far above prompt critical, there is not too much importance in the very serious cases attached to the exact value of the effective delayed neutron fraction. This is not true for the less serious excursion.

The two-region systems have the lowest effective delayed neutron fractions. This results from the larger ratios of external to core volume. A further contributing factor to a low beta eff is the low value of the U-233 delayed neutron portion (.0026). Under some conditions beta eff may be as low as a few hundredths of a percent.

There have been a considerable number of calculations of the consequences of assumed nuclear accidents in the AHR and, to a lesser extent, in the LMFR and MSR. These calculations, done on particular designs, have for the most part, indicated consequences that are not too serious.

However, in none of the concepts can the negative temperature coefficient of reactivity be counted on as a completely certain protective device against dangerous reactivity (and hence temperature) excursions.

Because of the many variations of the systems under consideration, it is difficult to make any generalizations as to the relative ratings of the three concepts on their susceptibility to nuclear accidents.

3. Chemical Accidents

As in other reactor systems, there are possibilities of various chemical reactions in the fluid fuel systems. Some of these are:

a. Aqueous Homogeneous Reactor

(1) A radiolytic gas explosion hazard exists, contributed to by possible failure of the catalytic recombiner.

(2) Zr-D_2O reaction - A reaction between the core tank wall material and water is, in principle, possible. Conceivably it could be instigated by the two-liquid phase separation or uranium-rich solid deposition, and consequent formation of a hot spot and melting through of a hole in the core tank. The design provided should make this less likely than was the case in HRE-2, where a melt-through occurred (without major reaction). In the event it does occur, the probability of a serious metal-water reaction is unknown.

b. Molten Salt Reactor

(1) Stored energy in graphite - This probably is not a serious consideration due to the high operating temperature.

(2) Molten salt reactions with INOR-8. This has been indicated to require quite high temperatures and, hence, unlikely as long as the fuel is uniformly distributed (dissolved).

(3) Molten salt reaction with water does not liberate a large quantity of energy, although heating the water could create pressure.

c. Liquid Metal Fuel Reactor

(1) Stored energy in graphite (same as MSR).

(2) Na-H_2O reaction, with liberation of hydrogen and heat.

(3) Contact of bismuth with sodium or container materials should not liberate much heat; reaction with air is slow at conceivable reactor temperatures.

4. Primary System Integrity Failure

The most characteristic serious accident in the fluid fuel systems is a failure in the integrity of the primary system.

Maintaining the integrity of the primary system in the fluid fuel systems can be made more difficult by reaction between the fluids and their containers. As an example there can be serious effects due to mass transfer in the LMFR, stress corrosion cracking in the AHR. No indication of similar risk has been found for molten fluorides in INOR-8.

In the AHR, the consequences of any break or leak are far more serious, since the spread of the fission products will be more widespread.

5. Containment

Among the systems under consideration, the AHR has proposed a large steel sphere or sealed cell for containment while the two non-aqueous systems have proposed closer fitting containment systems. The sealed cell of the AHR is designed to the requirement that it contain the pressure produced by accidental discharge of the fluids of the reactor system. The required volume within the containment

systems for the other two cases is not so great, since the vapor pressures of the fluids are very low.

It is not deemed possible to assign a higher probability of penetration of active material (to the outside of the containment system) in the event of breaching the primary system, for one reactor type over the probability for the others.

Consideration of the maximum credible accident for each of these systems ultimately depends on the question of how large a fraction of the fission products (or other toxic materials, e.g. Po^{210} in LMFR or T_2 in MSR) in the system might pessimistically be assumed to be released outdoors in an accident. Such considerations thus are strongly dependent on which variation of each of the systems is being considered. Some may have continuous removal of fission products while others let them accumulate for say, 20 years.

In general, however, due to the presence of high pressure, the AHR would have to be assigned a larger "maximum credible" accident than would the non-aqueous system. In the event its outer containment is breached, the fission products would then constitute a public hazard. If the close fitting containment of the non-aqueous systems is breached, there still remain several relatively leaktight barriers. Furthermore, the carrier would freeze, imprisoning the fission products.

G. Chemical Reprocessing *

The liquid fuel reactor systems are distinguished from other reactor types in having most of the facilities -- perhaps all the facilities -- found in a typical radiochemical processing plant. Numerous samples of radioactive fluids must be withdrawn, transported, and analyzed to follow and control plant operations. Radioactive fission gases must be treated, aged, sampled, analyzed, and in some instances bottled and shipped off site.

There are other points of similarity to radiochemical processing plants. Equipment will have to be removed from contaminated areas, decontaminated and repaired, or decontaminated and buried on the site. These operations require waste facilities which are similar to, although much less extensive than the facilities found at a radiochemical processing plant.

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* For divergent views, see project statements, pages 76, 95, and 169.

1. Minimum Reprocessing Requirement

For two of the three reactor concepts, it has been calculated that minimum fuel costs would be obtained with long cycle times, up to 30 years, without any reprocessing except for removal of fission gases. U-235 would be added continuously to compensate for poison build-up and depletion of fissile inventory.

It is anticipated that in actual practice some fraction of the fuel will become contaminated with impurities caused by such things as air and water vapor inleakage, less than perfect purging of the system or sealing of equipment, corrosion products, fission products, radiation damage, change in slurry characteristics, etc. The actual extent to which build-up in these systems may occur will not become known until a reactor experiment is operated for an extended period of time. In the meantime, it seems prudent to assume that impurities will form to a minimum extent of 5% of the total inventory, i.e. the entire inventory will be contaminated once in 20 years. In addition, it is felt that no utility would be willing to build and operate a fluid fuel reactor plant without a means of cleaning up the fuel should it become contaminated, either by gradual degradation or by accidental means.

The reprocessing plants provided in the reference design meet, or exceed the 5% limitation described above.

2. Off Site vs. On Site Reprocessing *

For chemical reprocessing of the fuel, there is a choice of doing this either off site or on site. Off site reprocessing involves many problems that are not generally recognized.

In order to process the high melting fuels (liquid metal or molten salt) off site, the reactor operator must can his fuel in sealed, dissoluble cans. In addition, the cans must be small, preferably no more than 2" in diameter and 4' long, or constructed in such a manner to aid dissolution; otherwise the capacity of the very expensive dissolution equipment will be seriously reduced. These cans must also be capped, seal-welded, inspected, decontaminated, handled remotely much as solid fuel elements are handled, and shipped to a processor in heavily shielded casks. When the produce is returned by the processor, some months later, it must be returned in shielded casks. The one operation of fuel canning for shipment off site could require about the same investment and involve about the same annual expense as processing the fuel on site.

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* For divergent views, see project statements, pages 76, 95, and 169.

It also should be noted that any centralized plant will hold a customer's material for a year or perhaps several years in order to accumulate enough fuel for a production run. When a sufficiently large inventory is accumulated, the fuel will be processed as consecutive batches after flushing out the plant. It is only in this fashion that the fluid fuel operator can be certain of receiving his product back uncontaminated by tramp diluents, or if material is to be sold, of receiving proper credit for the valuable isotopes.

The problems involved in shipping aqueous fluid fuels off site are complex also. The fuel must first be evaporated, calcined, and probably vacuum purged to recover substantially all the contained heavy water. The operation must be done in special shipping containers or the residual oxides must be dissolved in strong acids or slurried in water and transferred to shipping containers. After several months cooling the shipping containers must be sealed, transferred to large casks and transported to the processing plant. Although shipments of dilute, neutralized fission product wastes have been made on a limited experimental scale, the transport across country of the daily effluent from a 1140 MWT aqueous homogeneous reactor plant poses questions of hazard and liability too involved to be considered further here.

A comparison of the equipment involved will show that the operator of a fluid fuels power plant has available on his site most of the auxiliary facilities required to process his fuel. When it is considered that the cost of a loading facility to can fuel for off site shipment appears to be as expensive as a facility to process the fuel on site, it becomes evident that on site processing of fluid fuels is an economic necessity which is not likely to change with technological progress.

3. Reprocessing Concepts of Reference Reactors

a. AHR - Thorex Process

Reprocessing of AHR fuel will be by the conventional thorex process, the technical feasibility of which has been adequately demonstrated by the actual operation of such a plant at Oak Ridge. Some additional equipment is required, however, to provide for the routine recovery in high yield and purity--particularly purity--of the heavy water associated with the fuel. If the purity of this recovered heavy water is not up to specification because of contamination with light water, considerable added expense will be incurred. The fact that this heavy water is heavily contaminated with tritium (DTO) contributes added problems of ventilation, personnel protection, protective clothing, tool decontamination, etc.

b. MSR - Fluoride Volatility Process

The most economical process for recovery of uranium from molten salt appears to be the fluoride volatility process. Although this process is simple in concept, it is more complicated than one might suspect. Corrosion is severe at the high temperatures involved, materials handling is complicated, batch fractionation and post treatments of the distillate UF_6 fractions are required to secure a satisfactory product. Also, the process involves a secondary high temperature fluidized bed reduction of gaseous UF_6 to solid UF_4 using hydrogen as reducing agent. The fluorination step appears to involve more cell decontamination work and personnel access problems than the other processes. Access is no problem when processing low irradiation level fuels. Access and maintenance in this application where the cell will handle kilogram quantities of fission products each day is a problem which may prove to be of considerable magnitude.

One major disadvantage of this volatility process in a nuclear breeding economy is that for the one-region reference reactor thorium is not recovered but is run to waste along with the isotopically separated lithium fuel diluent and the accompanying fission products. If thorium is to be recovered later, additional thorex facilities on a substantial scale will have to be provided.

c. LMFR - Thorex Process *

Although a flow sheet has not been developed the conventional thorex process probably can be used for processing the LMFR slurry fuel. The bismuth does have, however, significant effect on dissolving problems as well as requiring process dilution to hold solvent degradation within reasonable working limits. Furthermore, the bismuth cannot be recovered without the addition of further equipment.

It has been assumed that about 150 kg per day of Bi containing only about 4 kg of thorium, and about 200 grams of fissionable uranium, and up to about 0.6 kg of fission products will require facilities equivalent in cost to those required to process about 90 kg per day of normally irradiated thorium.

The equipment required for processing facilities which recover bismuth reflects the same maintenance-access problems involved in the volatility process except the problems probably are much worse due to aerosol generation and ruthenium volatilization which are to be expected. This process is totally undeveloped. It contemplates blowing bismuth with oxygen at elevated temperatures to slag and

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* For divergent view, see project statement on page 95.

float off oxides. Aqueous caustic may be used instead of oxygen but this will not mitigate the radioactive aerosol generation problem.

As a minimum figure, the recovery of bismuth can add $1 million investment (direct materials and labor basis) and may very well make the process unworkable. In view of the $193,000 annual saving realized by recovering bismuth, and the nebulous saving involved in shrinking the size of an already undersize thorex facility, there seems to be little incentive in pursuing this method.

d. Throughput Rates

The throughputs listed for each of the processing plants are based on an appraisal of the limiting factors involved, assuming no drastic shutdowns for cleanup or maintenance.

Throughput data for the well-developed thorex plant used on the aqueous homogeneous system is based primarily on Oak Ridge experience with this process. The yearly throughput rate of the reference design is slightly greater than the total system inventory and, therefore, well above the *minimum* reprocessing requirement mentioned above.

Throughput figures for the fluoride volatility process are based primarily on the experience of Argonne. The throughput rate of the reference design is 20% of the total system inventory.

Throughput figures for the LMFR bismuth-thorex (no bismuth recovery) process represent an educated guess as to the effect of bismuth on dissolving problems and the process dilution required to hold solvent degradation within reasonable working limits. The throughput rate of the reference design is based on 5% of the total system inventory.

The greatest uncertainty in throughput capabilities of these plants is not so much the name-plate throughput that can be squeezed through these small plants under the most favorable circumstances, but in the number of days per year such direct maintenance plants can be kept operating. Spills, breaks, or mishaps can, and occasionally do, put plants of this type out of commission for weeks or even months at a time if a bad spill occurs.

4. Fuel Reconstitution

An area of processing which has received little if any consideration in any of the three systems is the reconstitution of fuel from the thorium and uranium recovered by the thorex or other processes used. If recovered uranium and thorium are to be re-used

in these plants, a considerable investment will be required to prepare the appropriate thorium and uranium compounds to exacting reactor requirements under extremely high radiation conditions which characterize such recycled material. This tail-end processing, which may involve precipitation, calcination, particle size adjustment, recalcination, sample analysis, physical testing, and other steps, must be carried out completely remotely behind heavy shielding.

Although fuel reconstitution is common to all three reactor concepts, it is noteworthy that the product from the MSR fluoride volatility process only requires control over chemical purity where all other fluid concepts require close control over physical characteristics such as particle size, particle density, and even particle shape. For the MSR system, probably the only operations which might be required here would be the blending of diluent salt and makeup thorium fluoride with the recovered uranium fluoride product, standardizing the mixture chemically, and pelletizing the standardized blend. All these operations would have to be carried out behind shielding in a dry box filled with an inert argon or nitrogen atmosphere to avoid moisture absorption and subsequent hydrolysis of the fluorides (and precipitation of UO_2 in the reactor circuit).

A question which is asked is the reason why tail-end process shielding is necessary when working with freshly separated U-233 which is not particularly radioactive, the activity from U-232 only growing in after a prolonged time. Experience shows that complete decontamination of the processing cell and equipment is not feasible after each use. Particulate matter becomes distributed, equipment gradually becomes more radioactive as time goes on even though the batch being processed may be quite non-radioactive. Sooner or later, however, the entire cell and equipment comes to its equilibrium level of radiation intensity. It is at this time that the shielding is needed.

5. Investment Requirements

The capital investment requirements and estimated throughput capabilities for on site processing facilities for the reference plants are shown in the following table:

Fuel Cycle
Total Investment - Summary

Concept	AHR	MSR	LMFR
Process	Thorex	Fluoride Volatility	Bi-Thorex
Throughput	300 kg/day Th	60 kg/day salt	150 kg/day Bi
Bi recovered			no
Li recovered		no	
Th recovered	yes	no	yes
U recovered	yes	yes	yes
	Direct Materials & Labor		
Recovery process	4,330,000	3,600,000	2,600,000
Fuel reconstitution	1,100,000	100,000	1,100,000
Laboratory	700,000	700,000	700,000
Gas Facility	560,000	560,000	560,000
Stack	250,000	250,000	250,000
Xe, Kr recovery	1,000,000	1,000,000	1,000,000
Waste Facility	560,000	560,000	560,000
TOTAL M&L	8,500,000	6,770,000	6,770,000
Lumped Dist. Items @ 74%	6,290,000	5,010,000	5,010,000
Approx. Total Investment	14,790,000	11,780,000	11,780,000

The investment for the AHR process reflects a conventional thorex operation capable of handling the capacity shown. For the LMFR and MSR, these are considered minimum facilities that could be built and represent what are considered to be minimum investment costs. The higher investment for the MSR recovery process results from the fact that this part of the chemical processing is more complicated. Equipment required for MSR fuel reconstitution is, on the other hand, much simpler, as described above and therefore less costly. Bismuth could be recovered in the LMFR process but the value of the bismuth recovered would not pay for the added investment cost required.

6. Annual Operating Requirements

It has been found from experience that the annual operating

cost of large radiochemical processing installations amounts to 15% of the total full book investment. This does not include charges on capital and the costs of special materials (Li, Bi, D_2O). It does include all operating expense items such as supervision, salaries, wages, maintenance material, maintenance labor, general supplies, utilities, and appropriate associated distributive items.

Operating costs for small direct maintenance plants which are heavily weighted with hot laboratory facilities, skilled personnel, and substantial health physics requirements, amount to much more than 15%. In the absence of a better figure for the type operation considered here, it is suggested that the 15% factor be used.

VI. TECHNICAL FEASIBILITY OF BREEDING

In appraising the technical feasibility of breeding, the Task Force was requested to make a survey of available calculations and estimate the breeding ratios attainable in the various fluid fuel systems. In this estimate consideration was given to any fluid fuel reactor concept deemed to be a "reasonable extrapolation" of technology.

Also included are estimates of the additional cost to attain breeding, developments necessary to achieve specified amounts of breeding, and a discussion of the status of the basic physics data.

A. Breeding Potential

Since the need for breeding will in all probability arise, the question of the breeding potential of the fluid fuel system when operating on the thorium U-233 cycle is of great importance.

It is convenient to distinguish between the following systems:

Converters
"Hold own" breeders
Doublers

The converters encompass all systems with effective conversion ratios less than, say, 0.95.

The "hold own" breeders are systems with conversion ratios around unity. Their main feature is that they do not have sufficiently short doubling times to make considerations of that quantity of any significance. They do not, however, require any substantial amount of fuel after they are supplied with their initial inventory (which may be very large). In principle, such systems permit full utilization of nuclear fuel.

Doublers are breeders with doubling times short enough (under 20 years) so that reactor capacity based on bred fuel can expand with load growth.

Insofar as breeding is concerned, the major inherent difference between the three concepts under consideration is in the parasitic losses to the fuel carrier and to the graphite moderator in the cases where it is used. These losses can be minimized by a higher fuel

loading. If this is done in LMFR and MSR it tends to equalize the breeding gain in the three systems (but still leaving a large disparity in the doubling time unless LMFR and MSR go over to internally cooled systems).

In principle it is possible to obtain a breeder with a one-region system, if it is sufficiently large. It is possible that future central station requirements might favor power outputs corresponding to such large sizes. Even then, however, it is probable that it would be advantageous to satisfy this larger power requirement with a two-region system.

The following "optimum" breeders are characteristic of the best breeders that can reasonably be expected to be obtained in the order of 25 years. The feasibility of these systems is dependent on the successful solution of problems listed later.

The optimum breeders for each of the concepts have the following characteristics:

AHR

The AHR optimum breeder is a two-region system with a solution core, slurry blanket with breeding in the blanket only. There would be continuous Xe removal, limiting the Xe loss to approximately .01 per neutron absorbed in fissionable material. Internal recombination of the radiolytic gases woulc be used provided a suitable catalyst can be developed which will not result in too large a parasitic loss. Otherwise, external recombination will be used. The neutron leakage would be limited to approximately .03. There would be frequent processing for removal of the non-volatile fission products and corrosion products. It is expected that the losses to the non-volatile fission products and corrosion products would be in the range .03 to .06. The frequency of processing would be determined by a compromise between cost against the doubling time. A breeding ratio of approximately 1.07 to 1.10 is expected. The total fuel inventory in both the reactor and processing for 1140 thermal MW(333 MWE gross) would be approximately 600 kg. The corresponding doubling time would be 15 to 20 years.

MSR

The MSR optimum breeder is a two-region system with a graphite structure in the core but not in the blanket. It has a heavily loaded core. Breeding is accomplished in both the core and blanket. There would be continuous Xe removal. The

neutron leakage would be limited to approximately 0.03. There would be frequent processing for removal of the non-volatile fission products.

A breeding ratio of approximately 1.05 is expected at a specific power of approximately 1 thermal MW per kilogram of fuel. Preliminary studies have been on a 100 thermal MW system. It is assumed that the salt is retained after processing so that the Li^6 is substantially burned out.

LMFR

The LMFR optimum breeder is a two-region system. It has a graphite structure in both core and blanket. It uses a solution for the core and slurry in the blanket. There is breeding in the blanket only. There is continuous removal of Xe. The neutron leakage would be limited to 0.03. There would be frequent processing for removal of non-volatile fission products.

A breeding ratio of approximately 1.05 could probably be obtained at a specific power of approximately 1 thermal MW per kilogram of fuel. Preliminary studies have been on a 800 thermal MW system.

Depending on the successful solution of certain technical feasibility questions, the AHR has a chance of having a reasonable doubling time. On the other hand, the present concepts for MSR and LMFR, although they can lead to "hold own" breeders, have no possibility of being good doublers. The only possibility for MSR and LMFR to be good doublers would be by the internal cooling concept. This, however, introduces a new development program with many difficult and complicating features. If solved successfully the potential in terms of doubling time would then be roughly equivalent to that of the presently envisaged AHR.

In evaluating the three concepts as "hold own" breeders, the difference between the systems is not large. The AHR has a breeding ratio advantage of about .04 over the non-aqueous systems. In all probability the breeding potential of the two non-aqueous systems are very close. The higher parasitic capture of the molten salt carrier relative to bismuth is compensated by the smaller fluid volumes (both in the core and in the external system) needed in the molten salt case. At this time it is not possible to make a relative rating on the breeding potential of the two non-aqueous systems. In both of these systems the breeding ratio, when around unity, can be improved slightly at the expense of fuel inventory. The molten salt reactor has a higher solubility limit and is consequently more

flexible in this respect. In any event, when operating on a "hold own" basis, there is probably little advantage to push the breeding gain up slightly at the required cost in fuel inventory.

B. Additional Cost to Attain Breeding

Present design studies lead to the conclusion that converter systems are now the most economival. It is of interest then to estimate the additional cost to make these systems breeders. In the case of the AHR with 15 year doubling time, the main additional cost above that of the same design operating in the most economic fashion is the cost involved in processing more frequently. A rough estimate of the total additional cost (including fixed charges, operation and maintenance, and fuel and processing costs for AHR as a 15 year doubler compared to the AHR minimum cost reactor) is approximately 1 mill/kwh. For the LMFR and MSR there are two separate cases to consider. First, there are the modifications of these systems to make them "hold own" breeders. The cost of this above that of the LMFR and MSR reference designs is of the order of 1 mill/kwh. The second case is the far greater extrapolation to internally cooled systems so that they operate as doublers. It is essentially impossible to make any estimate of the additional cost involved since at this stage comparatively little is known of the technical feasibility of these systems. Among the problems to be solved are the development of adequately impermeable graphite (or possibly beryllium) tubing and extremely reliable graphite-to-metal seals.

C. Necessary Developments

The main questions associated with the technical feasibility of breeding (more specifically, of obtaining a reasonable doubling time) in the AHR are:

1. removal of Xe - slurry core; solution core
2. removal of non-volatile fission products and corrosion products
3. core tank reliability
4. economic reprocessing
5. stability of solution and slurries under concentrations needed for breeding.

The main questions associated with the technical feasibility of breeding in the non-aqueous system on a "hold own" breeder basis are:

1. removal of Xe
2. removal of non-volatile fission products (and corrosion products if non-negligible

3. feasibility of adequately blanketing systems on top and bottom
4. feasibility of graphite or possibly beryllium as the core tank material
5. core graphite reliability
6. economic reprocessing
7. stability of solutions and slurries under concentrations needed for breeding.

The main feasibility question with respect to the non-aqueous systems as good doublers is that of developing an internally cooled system.

D. Status of Basic Physics Data

Due to uncertainties in some of the basic data, primarily eta (number of neutrons emitted per absorption) of U-233, precise prediction of the breeding potential is uncertain. The status of the more important cross section data leading to uncertainties is as follows:

The current best estimate for the value of eta of U-233 at thermal energy is 2.28. There are a number of measurements centered around this value. There are, in addition, some British measurements yielding a substantially lower value of 2.18. The conclusions in this evaluation are based on a value of 2.28 in the belief that this is the most probable value. It, of course, must be borne in mind that the estimates of breeding ratio and, more importantly, doubling time are sensitive to uncertainties in this quantity.

Not only the thermal value of eta of U-233 is important but also its energy variation in the epithermal region. This is particularly true with reference to those variations of MSR and LMFR which have substantial amounts of epithermal fissions.

It seems reasonable to assume an uncertainty of the order of $\pm$ 0.05 in the value of eta both in the thermal and epithermal range, notwithstanding the fact that this uncertainty does not overlap the low British thermal value. A corresponding uncertainty in the breeding ratio of $\pm$ 0.05 results.

Even with the uncertainty in the eta value, the conclusion that all the systems can be made to be "hold own" breeders is probably safe. This follows from the fact that breeding gains based on the most probable values are estimated to be somewhat over unity. Furthermore, within the meaning of "hold own" breeder, a value slightly less than unity is essentially as good.

There is considerable uncertainty in the long-term fission product cross sections. This is of great importance in the long burnup, low conversion systems. However, in the systems optimized as breeders, the uncertainty is not too important; the main effect is then in determining the necessary frequency of reprocessing.

There is also considerable uncertainty in the Pa-233 absorption cross section, particularly in its dependence on pile spectrum. However, it is not too important in any of the optimum breeder systems since in none of them are the Pa-233 losses large.

VII. POWER COSTS

A. Preparation of the Comparative Power Cost Tabulations

1. Purpose

The cost estimates were prepared to provide a basis for comparing the relative power cost potentialities of the molten salt, liquid metal fuel, and aqueous homogeneous reactor concepts for large central power plants, assuming them to be developed in accordance with the reference designs submitted by the projects.

The AEC assignment to the Task Force did not request comparison with estimates for any other reactors, or for conventional plants. Emphasis was placed on estimating the relative costs, to reflect as nearly as possible the essential differences between the concepts; there was less emphasis on determining absolute cost levels. Hence, the plant and power cost estimates should not be considered as being comparable on an investment or mills/kwh basis with cost estimates for other plants or reactor concepts, and the Task Force study does not attempt to evaluate the relative merits of these and other reactor types or conventional power plants.

2. Time Perspective

The tabulations are not intended to represent the costs of building fluid fuel power reactors now; neither do they predict the cost of building a first plant at some future date. It is assumed that when the plants are built, concepts and designs will have been fully developed; that experience will have been obtained in the operation of a prototype and at least one large-scale plant; and that components will be obtainable from commercial sources.

On the other hand, the estimates are not considered as reflecting the long-term potentialities of the concepts. They do not take

into account economies that might be expected from larger plant capacities, larger on site chemical process operations, and more economical chemical processes.

3. Procedures Followed in Making the Estimates

The comparative estimates were based on a review of the concepts for which conceptual designs and cost estimates were submitted by the respective projects. Comparative plant investment and power cost estimates were prepared by the Task Force on the basis of these data, taking into account judgments of Task Force members in relation to their several specialized technical fields.

Since the designs and cost estimates submitted were in different stages of development and had been prepared by different groups employing various design assumptions and sources of data, it was considered necessary to adjust them to a consistent cost estimating basis. The Task Force considered that some features of the plants were common to all three concepts, i.e. that the costs of some features did not depend upon the characteristics of the particular reactors; and that for other features in the present stage of development there was no way of evaluating the differences. For plant items in these categories, identical costs were assumed for all three concepts.

For most of the plant items, essential cost differences were identifiable. These items included the reactor buildings, reactor, heat transfer loops, steam generators, power equipment, and many of the auxiliaries. Calculations were made to check the reasonableness of the design assumptions and cost estimates that had been presented. Data were adjusted to a common basis, in order to minimize the effect of non-essential variations. The estimates were checked against detailed estimates for conventional plants, against the conceptual plant description, and in relation to each other in an effort to correct omissions and assure reasonable comparability. The effects on construction costs of unusual construction materials such as INOR-8, Zircaloy, and graphite, and unusual requirements for leak-tightness, reliability, and accessibility for remote maintenance were considered. Efforts were made to reconcile the widely-varying judgments of the Task Force members regarding cost estimates in fields where experience on fluid fuel reactors is limited and sometimes totally lacking. (Examples are remote maintenance facilities and spare parts.)

The procedure in estimating fuel costs was somewhat different from the above, with less weight being given to the data submitted by the projects. The initial analysis submitted by the projects in regard to chemical processing were not considered by the Task Force

to be sufficiently developed for use in the overall cost comparisons. The Task Force adopted a general set of cost estimates for chemical plant investments, fixed charges, and operating costs covering the minimum on site processing facilities that were considered feasible for each of the three reactors. The cost picture thus evolved was considered by two of the project leaders (MSR and LMFR) to be inconsistent with the assumptions on which their design concepts were based. These project leaders have prepared new fuel cycle analyses based on fuel cycles and processes which they consider to show promise for future cost reduction. These alternative analyses are included in the respective project statements, pages 76 and 95.

Operating cost estimates for the power plants were based upon estimates submitted by the LMFR project for the operating personnel required for the various plant functions. Pay rates assigned were analogous to those in coal-fired plants. The annual maintenance costs were assumed to be 3% of the plant investment. This percentage is within the range of experience in maintaining various radiochemical facilities.

The annual costs of operating and maintaining the fuel processing facilities were taken as 15% of the investment in these facilities.

4. Ground Rules

In determining the annual fixed charges corresponding to the plant and inventory investment cost estimates, the following ground rules were applied:

Fixed Charge Rate for Depreciable Plant and Inventories	**14%**
This was arrived at as follows:	
Bond Interest (3.5% on 50% of capital	1.75
Preferred Stock Div. (5% on 15% of capital)	0.75
Common Stock (10% on 35% of capital)	3.50
Overall Return	6.00
Federal Income Tax (52/48 of return on common and preferred stock)	4.60
Other Taxes (real estate, etc.)	2.00
Insurance (other than 3rd party liability)	0.10
Sub-total	12.70
Depreciation (sinking fund @ 6% 30 yrs.)	1.30
Total Fixed Charges	14.00%

Thorium and fuel-carrier materials were considered as depreciable inventory.

Fuel Use Charge for Fissile Materials (applied to average inventories of U-235 and U-233 in the reactor during a complete fuel cycle)	4%
Plant Operating Factor (the equivalent of 7000 hours/year at full load - this is based on the very low incremental operating cost, consistent with the fuel cost estimates and adequate availability)	80%

B. Conclusions

A review of the overall summary of power costs, Table VII-1 which follows, shows significant differences in estimated cost among the three concepts in these major categories:

Reactor and Steam Generating Plant
Turbogenerator Plant
Chemical Processing and Waste Disposal Plant
Fuel Burnup, Use, and Inventory Charges

As can be seen from the detailed tables supporting the summary, these differences are accounted for primarily by the particular design features, and fuel cycle assumptions discussed below:

1. The high pressures and low temperatures in the primary system of aqueous homogeneous reactors are reflected in higher cost of the steam generators; and, because of the resultant lower conversion efficiency, in higher cost of the turbogenerator and power system components.

2. The indicated cost disadvantage of the molten salt system (compared to LMFR) results from the characteristics of the particular heat transfer system chosen for the reference design: the combination of a secondary fluid (barren salt) with a high-melting point and a Lieffler boiler system to permit a steam-cooled secondary heat exchanger. Not only does the Lieffler system increase the investment in equipment and buildings; it also requires development of a steam pump.

Other possible systems that have been proposed, such as one employing a low-melting-point salt in the secondary loop, are being investigated. If such a system proves feasible, there should be no great difference in the investment requirements between MSR and LMFR.

3. The aqueous homogeneous reactor reference design is for a breeder reactor; the other two are for converters. This difference

is reflected in (a) more frequent fuel processing for the homogeneous reactor, which requires a larger and hence more costly chemical plant; and (b) a net credit for fissile material produced, instead of a net cost for fuel burned. Further, the homogeneous reactor shows a higher fuel carrier investment because of the large volume of heavy water required. It is interesting to note that these fuel cost differences between the homogeneous breeder reactor and the non-breeder reactors approximately offset each other, with a slight cost advantage in favor of the breeder (for fuel processing investment and fuel cycle costs combined: 2.74 mills/kwh vs. 2.84 mills/kwh). This result obviously does not support the conclusion which has been sometimes expressed, that breeding necessarily incurs a fuel cost penalty. However, these results are not considered to support definitive conclusions, in view of the present state of development of the concepts, designs, and fuel processes.

Overall, the differences in power cost estimates among the three concepts are so small in relation to the probability accuracy of the estimates that they are not considered a determining factor.

C. Presentation of Cost Information

The overall summary of power costs, Table VII-1, which follows, is based upon more detailed information presented and discussed as noted below:

Descriptive Data	- Project Reports, Sections IX, X, and XI
Power Plant Investment	- Table VII-2 and notes
Chemical Reprocessing Plant Investment	- Section V-G
Summary of Fuel Costs	- Table VII-3 and notes
Chemical Reprocessing Operation and Maintenance	- Section V-G
Power Plant Operation and Maintenance	- Notes on Table VII-1 and Sections V-C and V-D

Table VII-1 — OVERALL SUMMARY OF POWER COSTS

The power plant investment tabulation is derived from incomplete conceptual designs and estimates submitted by the respective project leaders, adjusted to reflect experience-based judgments of Task Force Members, emphasizing relative costs based on essential differences between the concepts. Less emphasis has been placed on absolute costs or specific design details. It has been assumed that concepts and designs will have been fully developed, and that components will be commercially available at the time when the plants are built. The estimates are based on 1959 dollar values.

	REACTOR CONCEPTS								
	LMFR, SINGLE REGION, GRAPHITE MOD. Liquid Metal Fuel: UO_2-ThO_2 Slurry in Bi			MSR, SINGLE REGION, GRAPHITE MOD. Molten Salt Fuel: UF_4-ThF_4-BeF_2-LiF			AHR[f] TWO REGION, D_2O MODERATED Aqueous Homogeneous: UO_2SO_4 in D_2O with blanket of ThO_2 in D_2O		
DESCRIPTIVE DATA									
Heat Output, MW	825			760			1143		
Electrical Output, MW - Gross	333			333			333		
- Net	312			318			316		
Net Station Efficiency, percent	38%			42			28%		
Fuel Inventory, Kg: U-235 (Initial)	770			829			774		
Thorium	23,000			38,400			123,000		
Reactor Operating Pressure	150 psig max.			75 psig			2000 psig		
Reactor Operating Temp. Range, °F	750 - 1050 F			1075 - 1225 F			482 - 554 F (core)		
Steam Conditions	2000 psia; 1000 F			2000 psia; 1000/1000 F			400 psia; 445 F		
POWER PLANT INVESTMENT[a] (Fixed Charge Rate @ 11%)	$ Millions Invest.	Annual Cost	Mills per kwh	$ Millions Invest.	Annual Cost	Mills per kwh	$ Millions Invest.	Annual Cost	Mills per kwh
10 Land	0.5	.07	.03	0.5	.07	.03	0.5	.07	.03
11 Structures & Improvements	9.2	1.28	.55	9.7	1.35	.58	8.8	1.24	.53
12 Reactor & Steam Generating Plant	22.2	3.11	1.33	26.0	3.64	1.56	29.0	4.05	1.74
14 Turbine-Generator Plant	16.8	2.35	1.01	16.4	2.30	0.99	20.5	2.86	1.22
15 Accessory Electrical Equipment	1.8	.25	0.10	1.3	0.19	0.08	1.2	0.17	0.08
16 Misc. Power Plant Equipment	0.2	.03	0.01	0.2	0.03	0.01	0.2	0.03	0.01
TOTAL DIRECT COSTS - POWER PLANT	50.7	7.09	3.03	54.1	7.58	3.25	60.2	8.42	3.61
Distributives (Indirect, Overhead, & Contingencies) - 74%	37.5	5.25	2.26	40.0	5.61	2.40	44.5	6.24	2.67
Training & Preparation for Commercial Operation	2.0	.28	.12	2.0	.28	.12	2.0	.28	.12
Materials Inventories (Excl. Fuel)	5.2	0.73	.31	7.9	1.10	.47	5.0	0.70	.30
TOTAL POWER PLANT INVESTMENT	95.4	13.35	5.72	104.0	14.57	6.24	111.7	15.64	6.70
CHEM. PROCESSING & WASTE DISPOSAL PLANT[b] INVESTMENT 11%	11.8	1.65	.71	11.8	1.65	.71	14.8	2.07	.89
FUEL INVENTORY, USE, & BURNUP COSTS[c]		3.16	1.36		3.19	1.37		2.11	.90
		Annual Cost Details	Mills kwh Details		Annual Cost Details	Mills kwh Details		Annual Cost Details	Mills kwh Details
Thorium & Fuel Fluids Investment 11%	4.6	.65	0.28	2.6	.37	0.16	12.9	1.81	.78
Fuel Use Charge 4%		.60	.26		.72	.31		.52	.22
Fuel Burnup (and 6% s.f. to cover loss of value; also includes fuel losses in chem. processing)		1.69	.73		1.59	.69		-.49	-.21
Loss and Waste of Fuel Fluids (includes 6% s.f. for fuel recovery after 30 yrs.)		.22	.09		.51	.21		.27	.11
CHEM. PROCESSING PLANT OPER. & MAINT.[d] 15%		1.77	.76		1.77	.76		2.22	.96
POWER PLANT OPER. & MAINT.[e] (includes maint. @ 3% of Invest.)		3.40	1.46		3.67	1.57		3.88	1.66
TOTALS	111.8	23.33	10.01	118.4	24.85	10.65	139.4	25.92	11.11
TOTAL POWER COST-MILLS/KWH - Gross			10.0			10.7			11.1
- Net			10.7			11.1			11.5

NOTES:
- a/ For details and explanation of power plant investment data, see Table VII-2.
- b/ Chemical processing and waste disposal plant investment figures shown are based on the chemical processes and cost estimates presented in section V-G. For divergent views of the project directors on costs of chemical processing see pages 76, 95, and 169.
- c/ For details and explanation, Table VII-3.
- d/ The chemical processing plant operation and maintenance are computed as 15% of the chemical processing plant investment shown above, based on section V-G. For divergent views of projeet directors on costs of chemical processing see pages 76, 95 and 169.
- e/ Operating costs are based on estimates of personnel requirements and other expenses. Power plant maintenance has been estimated at 3% of total power plant investment, including distributive items.
- f/ Costs shown for AHR are for two region solution core power breeder. Lower costs estimated for slurry fueled reactor are presented on page 168.

Table VII-2

COMPARISON OF INDICATED LEVEL OF POWER PLANT INVESTMENT FOR REFERENCE CONCEPTS OF FUTURE STATIONS WITH FLUID FUEL REACTORS

Note: Figures below are not to be considered as reliable cost estimates. The figures represent adjustments to figures presented by project representatives in those categories where various disinterested task force members have experience and judge such adjustments as necessary. The figures represent investment levels judged to be somewhat in line with costs that might be obtainable in the distant future (in terms of 1959 dollars) and after all necessary research and development has been completed, all technical problems resolved (if possible) and after experience with design and operation of a few generations of full-scale power stations. It is judged impossible to provide any of these power plant concepts today for figures anywhere approaching these total levels.

PRODUCTION PLANT Output Capacity Description *		DATA: LMFR 333 MW $/KW		MSR 333 MW $/KW		AHR 333 MW $/KW	
310.	LAND & LAND RIGHTS		1.60		1.60		1.60
311.	STRUCTURES & IMPROVEMENTS		27.50		29.00		26.50
.1	Site Improvements	2.00		2.00		2.00	
.2	Site Facilities	2.60		2.60		2.60	
.3	Station Buildings	22.90		24.40		21.90	
.31	Reactor & Steam Gen Bldg	19.50		21.00		18.90	
.32	Turbine & Aux Bldg	1.90		1.90		2.40	
.33	Serv & Maint Bldg	1.00		1.00		1.00	
.34	Chem & Waste Bldgs	**		**		**	
.35	Other Station Bldgs	0.50		0.50		0.50	
312.	REACTOR & STEAM GEN. PLANT		66.70		77.90		86.80
.1	Reactor Equipment	6.10		5.00		11.20	
.2	Heat Transfer Equip	20.70		19.70		21.70	
.21a	Primry Sys Fil & Stg Equip	1.40		3.00		8.30	
.21b	Primry Sys Loop Equip	16.00		11.60		13.40	
.22a	Int Sys Fil & Stg Equip	.30		1.50		none	
.22b	Int Sys Loop Equip	3.00		3.60		none	
.3	Reactor Plant Auxiliaries	10.40		10.60		13.80	
.31	Radioactive Cont Equip	2.50		2.30		4.10	
.32	Let down & Recomb Equip	none		none		3.10	
.33	Reactor Off-gas Equip	3.70		3.70		3.70	
.34	Inert Gas Equip	0.30		0.30		0.30	
.35	Purge Equipment	0.20		0.30		0.90	
.36	Sys Heat,Cool,& Vent Equip	3.70		4.00		1.70	
.4	Steam Generator Equipment	4.00		16.00		18.00	
.5	Steam Generator Plant Aux.	5.40		4.70		2.10	
.51	Boiler Feed Water Equip	4.60		2.80		1.50	
.52	Feed Water Treating Equip	0.60		0.40		0.40	
.53	Serv Boiler & Fuel Oil Equip	0.20		1.50		0.20	
.6	Reactor & Stm Gen Pl Pipe Sys	5.10		6.70		4.80	
.7	Reactor & Stm Gen Pl Insul	0.50		0.70		0.50	
.8	Controls & Instr Equip	4.50		4.50		4.50	
.9	Hot Cell & Remote Maint Equip	10.00		10.00		10.00	
	WASTE & FUEL CLEANUP FACILITIES	**		**		**	
314.	TURBINE-GENERATOR PLANT		50.30		49.30		61.30
	Turbine Foundation	0.50		0.50		0.80	
	Turbine-Gen & Exciter Equip	37.50		37.50		38.30	
	Turbine Pl Aux (incl Crane)	0.60		0.60		0.60	
	Condenser & Auxiliaries	5.50		4.50		10.70	
	Turbine Pl Piping & Insul	0.20		0.20		0.20	
	Circulating Water System	6.00		6.00		10.70	
315.	ACCESSORY ELECTRICAL EQUIPMENT		5.30		4.00		3.70
	Foundations & Structures	0.20		0.20		0.20	
	Pwr & Conversion Equip	1.50		1.00		0.70	
	Conduits,Conductrs & Insuls	2.30		1.70		1.70	
	Switch, Cont & Protec Equip	1.10		0.90		0.90	
	Station Grounding System	0.20		0.20		0.20	
316.	MISC. POWER PLANT EQUIPMENT		0.60		0.60		0.60
	DIRECT CONSTRUCTION		152.00		162.40		180.50
	DISTRIBUTIVE EXPENSES						
	Indirect Const (15%)		22.80		24.40		27.10
	Overhead Const (30%)		45.60		48.70		54.20
	Allow for Omiss & Conting (29%)		44.10		47.10		52.30
	Exp Train & Prep for Reg Opr		6.00		6.00		6.00
	TOTAL CONSTRUCTION COST		270.50		288.60		320.16
	Spare Parts Inventory		15.00		15.00		15.00
	Heat Trans Fluids & Inert Gases		.60		8.60		.10
	TOTAL POWER PLANT INVESTMENT		286.10		311.20		335.20

* A listing of principal items considered to be included under each item in this table is contained in Appendix B.

** Investment for plant facilities to handle waste and reprocess fuel (i.e. Stack, Laboratory, Gas, fuel recovery, fuel reconstitution, Xe and Kr recovery and waste facilities) is not included in this Table but is shown in section V-G - Chemical Processing, page 31.

COMMENTS FOR TABLE VII-2

Certain elements of plant investment cost need not vary to any significant degree for purpose of relative evaluation of these reactor concepts, although such elements may vary considerably for specific locations and for specific installations for different utilities. For purpose of relative comparison of the plant investments, the concepts were all assumed to be suitable for some identical site and the following elements of investment cost were considered to be the same for all of the concepts:

Land and Land Rights	Turbine Plant Auxiliaries
Site Improvements	Turbine Plant Piping Systems
Site (General Use) Facilities	Access. Elec. Equip. Fdns. & Structures
Service & Maintenance Bldgs.	Station Grounding Systems
Misc. Station Bldgs.	Misc. Power Plant Equipment

Other elements of plant investment cost concerning which little or no detail variances could be ascertained because of lack of specific ideas having been developed pertaining thereto, but for which provisions rather common to each of the concepts must be made, were considered to be the same for all of the concepts. These are as follows:

Reactor Off-Gas Equip.	Control & Instru. Equip.
Inert Gas Equipment	Hot Cell & Remote Maint. Equip.

Elements of plant investment cost where distinguishing differences are judged to exist and brief explanations of these differences are as follows:

a. Nuclear Steam Generator Building: Figures presented by the project representatives are considered to be of proper order for the AHR and MSR concepts. Building for the MSR and LMFR are considered essentially the same, except for additional requirements of the MSR concept essential for extra loop system in the heat transfer facilities between the reactor and the superheated-reheated steam system.

b. Turbine and Auxiliaries Building: Figures used for the LMFR and MSR concepts are considered to be essentially equivalent and similar to cost for this element taken from a fossil fuel plant having like turbine and auxiliaries characteristics. Additional space required due to much greater condensing capacity and related much larger size of low pressure, saturated steam, non-reheat, tandem compound, double flow turbine-generator equipment required for the AHR accounts for the difference reflected in the figures for this element.

c. Reactor Equipment: Figures for all systems were developed from material requirements and weights for the respective specifications presented, considering graphite internals for both the LMFR and MSR concepts and core vessel and thermal shield internals for the AHR concept. Also considering control rods and operating mechanisms for the LMFR and MSR concepts and supports and miscellaneous hardware for all of the vessels.

d. Heat Transfer Equipment and Reactor Plant Auxiliaries: Figures presented by project representatives were used except in those cases where disinterested task force members had experience and basis for judging that adjustments were necessary. Also, figures judged to be reasonably representative were inserted for elements of cost where omissions were evident. In order to insure reasonably complete coverage and as much accuracy as possible with respect to the necessary principal equipment requirements for the heat transfer and reactor plant auxiliaries, these features of the plant equipment requirements were broken down into approximately 100 items for the investment cost evaluation.

e. Steam Generator Equipment and Steam Generator Plant Auxiliaries: Figures presented by project representatives were used for the steam generator equipment. However, figures for all systems were developed for feedwater heaters, boiler feed pumps, deaerator and auxiliaries, condensate and heater drain pumps, condensate storage tanks, reducing and desuperheating station (except for the AHR concept, requiring none), and feedwater treating equipment. The figures developed for these elements reflect the specific differences set by the requirements of the steam generator characteristics of the respective concepts.

f. Reactor Plant Piping and Insulation: Figure presented by the project representatives for the LMFR concept was checked and used, and approximate weights of piping were ascertained for all of the concepts and figures were developed for the MSR and AHR concepts by factoring with relation to the LMFR figure. Figures for insulation were judged to be equivalent to 10% of the piping figures used.

g. Turbine Plant Foundations and Turbo Generator Equipment: Figures for the LMFR and MSR concepts are considered to be essentially equivalent and similar to cost for this element taken from a fossil fuel plant having like turbine characteristics. Higher figures for the AHR concept are due to much larger size condensing capacity and related much larger size of the low pressure, saturated steam, non-reheat, tandem compound, double flow turbine-generator equipment and the much greater size condensing capacity required for the AHR concept. Turbo generator equipment costs for each of the concepts were taken from current catalog prices to which was added an amount of 10% for installation.

h. Condenser and Auxiliaries: Figures for all systems were developed based upon relative difference in requirements for condensing capacity set by requirements of the characteristics of the respective concepts.

i. Circulating Water System: Figures used for the LMFR and MSR concept are considered to be essentially equivalent and similar to cost for this element taken from a fossil fuel plant having like condensing water requirements. Requirements for additional condensing water due to characteristics of the AHR concept accounts for difference reflected in the figures for this element.

j. Power and Conversion Equipment, Conduits, Conductors and Insulators, Switching, Control and Protective Equipment: Figures for all systems were developed from an estimate, based on specifications presented by project representatives, of horsepower for plant auxiliary equipment and factoring with relation to requirements of a fossil fuel plant of the same comparable output capacity.

k. Distributive Expenses: Indirect construction costs were judged to be 15% of Direct Construction Cost and Overhead Construction Costs were judged to be 30% of Direct Construction Cost based upon experience of Architect Engineer representatives building fossil fuel installations. An allowance for Ommissions and Contingencies of 29% of Direct Construction Cost was judged to be a minimum prudent allowance considering the status of the art at this time.

Table VII-3
SUMMARY OF FUEL COSTS
FOR THE THREE CONCEPTS, BASED ON 333e MW (Gross)

	LMFR			MSR			AHR		
	$M Capital Cost	$M Annual Cost	Mills per kwh	$M Capital Cost	$M Annual Cost	Mills per kwh	$M Capital Cost	$M Annual Cost	Mills per kwh
1. Carrier Inventory (14%)	3,865	541	.232	1,533	213	.092	9,990	1,400	.601
2. Thorium Inventory (14%) ($22/kg)	506	71	.030	845	118	.051	2,700	378	.162
3. Prep of Initial Fuel Chg (14%)	250	35	.015	250	35	.016	250	35	.015
Sub-Totals (1,2,3)	4,621	647	.277	2,628	366	.159	12,940	1,810	.778
4. U-Fuel Inventory @ 4%	(15,000)	600	.258	(18,000)	724	.311	(13,140)	525	.225
5. Fuel Burn-Up		1,670	.717		1,570	.674		-504	-.216
6. Sinking Fund for Fuel Dep		15	.007		15	.007			
7. Thorium Burn-up		5	.002		5	.002		9	.004
8. Losses in Chem Process			.003			.013			(Inc. in Fuel Burn-up)
Sub-Totals (5,6,7,8)		1,690	.729		1,590	.696		-495	-.212
9. Chem Proc(Opr & Maint Cost)	(11,790)	1,770	.758	(11,790)	1,770	.758	(14,790)	2,220	.952
10. D_2O Loss (2½%)								250	.107
11. Bismuth Loss (5%)		193	.083						
12. Salt Loss (20%)					306	.131			
13. Th Loss (20%)					169	.073			
14. Sinking Fund (6%) for recovery of U at end of 30 years		26	.011		32	.014		20	.009
Sub-Totals (10,11,12,13,14)		219	.094		507	.218		270	.116
Annual Cost ($M)		4,926			4,957			4,339	
Fuel Cost (per kwh gross output)			2.116			2.142			1,859
Fuel Cost (per kwh net output)			2.26			2.24			1.96

The fuel cost figures in Table III are based on information supplied by the Project Directors for the reference design reactors. On-site reprocessing amounts to approximately 140% of total fuel inventory per year for the AHR, 5% per year for LMFR, and 20% per year for MSR. The costs shown for the AHR are for the two-region solution core slurry blanket concept. An explanation for some of the figures in the Table is given below:

Item No.
in Table

1. The carrier costs (bismuth) for the LMFR are based on a total fluid fuel inventory of 2800 cu. ft.

2. The capital cost for the carrier for the MSR is based on the total cost of $1700 per cu. ft. including the cost of the Li^7. The major fraction of this cost is for purification of the salt.

3. A $250,000 charge was assumed for each concept for the preparation of the initial fuel charge. There are no good data on what this initial cost would be.

4. The fuel inventory use charges are based on a value of $17.11 per gram, for U-235 and $15.00 for U-233.

5. In the AHR a credit is shown for the excess U-233 that is produced.

6. To pay for the reduction in value of the U-235 (because it is mixed with other isotopes) when the fuel charge is finally processed at the end of thirty years, a 6% sinking fund is used.

9. The operating cost of the chemical plant is considered a direct fuel cost. This operating cost is taken as 15% of the total capital investment in chemical plant. The capital cost charges for this plant are included under investment costs.

11. Five percent of the LMFR fuel is chemically processed each year without recovery of the bismuth. If bismuth were to be recovered the chemical plant would require an estimated $1,740,000 additional investment.

12.) The MSR uses the fluoride volatility process for processing 20%
13.) of the fuel inventory each year with no recovery of either the salt or the thorium.

14. At the end of thirty years each reactor has a full charge that requires reprocessing. For the AHR it is assumed that the chemical plant would operate for 250 days at which time the full charge would be completely processed. The inventory charge is applied only to the U-235 and U-233 for this period since the D_2O and thorium have already been fully depreciated. For the LMFR and MSR it has been assumed that it would be possible to package and ship off-site the entire full charge within a six-month period. The inventory charge would be for one year. Cost of shipping and processing at an off-site plant is assumed to be $22 per kg. Fuel burn-up costs are based on the assumption that the U-235 burned will be charged for as current operating costs. This quantity is based upon the difference between the cumulative total of U-235 fed into the reactor, and the calculated residual quantity of U-235 and U-233 in the reactor at any particular time. This residual quantity is accumulated for as fissile material inventory, against which the 4% fuel use charge is levied.

VIII. RESEARCH AND DEVELOPMENT PROGRAMS

The accompanying Table VIII-1 summarizes time schedules and funds required for development programs as presented by the Project Directors in Sections IX, X, and XI. These estimates indicate their appraisal of the effort required to develop the state of the art for each concept to a level suitable for designing and constructing commercial power plants. The Task Force has not considered these programs in detail nor put them on a comparable basis.

To carry through to a practical design of a commercial power plant, each program requires research and development, design, construction, and operation of a reactor experiment and of a prototype reactor. The cost of the entire development program for any one concept would be something above one hundred million dollars. Ten years would be required. There is no indication that any one would be substantially less costly than the others.

All of these estimates are, of course, based on the assumption that the problems on which technical feasibility depends will be successfully resolved. The programs should be reviewed periodically to appraise their progress.

The Task Force believes that in these programs major emphasis should be placed on extensive design and cost studies and evaluation of indirect maintenance schemes suitable for large-size radioactive equipment. Subsequent research and development, reactor experiments, and prototype operation should be guided by the results of these studies.

In the event that strong emphasis in the Commission's development program is placed on breeder systems: the AHR program would be essentially unchanged; the MSR and LMFR would place greater emphasis on the development of two-region graphite moderated reactors and on reprocessing systems.

TABLE VIII-1

Time and Cost Estimates
(Project Directors appraisal as presented in Sections IX, X, and XI)

	AHR			MSR			LMFR		
	Start	End	MM$ Cost	Start	End	MM$ Cost	Start	End	MM$ Cost
R&D for Reactor Experiment	-	7-63	18	-	7-63	19	-	4-62	14
Design, Construction, & Operation of Reactor Experiment	7-61	7-66	15	7-59	12-65	18	-	9-64	14
R&D for Prototype	1962	1967	35	1963	1968	35	-	1964	17
Design, Construction, & Operation of Prototype	1964	1969	59	1965	-	50	1963	1971	45
R&D for Full-Scale Plant	1968	1971	5	1968	1971	Not Given	1965	1969	10+
			132			122(+?)			100 +

IX. MOLTEN SALT REACTOR

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IX - MOLTEN SALT REACTOR

A. The Objectives of the Molten Salt Reactor Program

1. Low-Cost Power

Assuming success of an orderly development program it should be possible to start construction of a 333,000 KWE electrical gross plant of the type discussed herein with these power costs:

	Mills/Kwh Gross
Power Plant Investment ($311/KWE)	6.24
Chemical Processing Plant Investment ($35/KWE)	.71*
Fuel Inventory, Use Charges and Burnup	1.37
Chemical Processing:Operation & Maintenance	.76*
Power Plant Operation & Maintenance	1.57
Total Power Cost - Gross	10.7
Total Power Cost - Net	11.1

With reasonable developments and design refinements, the capital cost could be reduced to perhaps $304/KW. In the opinion of the project director, the fuel cost could be reduced to 1.7 mills/KWH by off-site processing*, including fuel charges and chemical processing operation and maintenance.

2. Breeding

Present conceptual designs would be break-even breeders after equilibrium uranium isotopic balances are attained. These designs include a two-region homogeneous reactor and a single-region graphite moderated reactor with a 20' diameter cylindrical core.

A two-region externally cooled graphite moderated breeder with a graphite core shell would have a conversion ratio of 1.05 after an equilibrium distribution of uranium isotopes is attained. This yields a doubling time of 44 years on the reactor design postulated. It is possible that with improvements in means of power removal, shorter doubling times are attainable. The two-region homogeneous break-even

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* See page 76.

breeder, after it is supplied with U-233, would have power costs about equal to those listed for the economic reactor.

The one-region break-even breeder might also have no higher power costs than the economic reactor. It does, however, require development of a salt purification process that can handle the entire charge of salt about once a month.

No estimate has been made for the two-region graphite moderated breeder.

B. The Approach

1. Summary of Concept

a. Basic Reactor Concept

The Molten Salt Reactors utilize molten fluoride salts as the solvents for both the fuel and the fertile material. The fluoride salts themselves have about half the slowing down power of graphite, and so reactors may be homogeneous with only self moderation by the salt, or may use graphite as the principal moderator. The container material, a nickel-molybdenum alloy, is compatible with both the salt and graphite. A large variety of reactor types can be constructed using these basic materials of construction and utilizing U-233, U-235 or Pu fuel and Th or U-238 as a fertile material.

b. Low Cost Power Reactors

(1) For low cost power, a single region externally cooled graphite moderated reactor utilizing thorium as the fertile material and U-235 as the added fuel is being given primary consideration. U-238 can be used as a fertile material in the same reactor at slightly higher power costs.

(2) Another reactor type given considerable study is a two-region externally cooled homogeneous reactor. This reactor type avoids the complication of a graphite moderator, but has a lower conversion ratio and consequently a slightly higher fuel cycle cost.

c. Breeding Reactors

Three types of molten salt reactors are considered for breeding. In order of increasing difficulty of construction, they are: the two-region externally cooled homogeneous reactor, the one-region externally cooled graphite moderated reactor, and the two-region externally cooled graphite moderated reactor. The first of these, relatively

easy to construct, will just barely breed, as will the one-region graphite moderated reactor. The two-region graphite moderated reactor is better as a breeder but has a doubling time of about 40 years.

d. The Special Merits of the Molten Salt Concept

The molten salt reactor is the only one of the three fluid fuel reactors that does not require a slurry. The use of solutions for both fuel and fertile material simplifies mechanical design, avoids erosion problems, simplifies some steps in chemical processing, reduces concern over nuclear instabilities, avoids settling or drainage problems that may occur when a fuel pump stops, and decreases the required amount of development work by a large factor.

Corrosion is not a problem in the molten salt reactor system for temperatures up to 1300°F. The basic mechanism of the corrosion that does occur at higher temperatures is well understood; the stability of the INOR-8 alloy to fluoride salts does not depend on a surface film or on additives.

The state of the materials and basic component development program is such that a design and development program for an experimental reactor reactor could be inaugurated at once. The ultimate economic potential of the system can be properly appraised within a ten-year period, so that this reactor, if successful, would join the class of reactors "achieving economic power within ten years" as defined by the Ad Hoc Committee.

The wide range of solubility of U, Th and Pu in the fluoride salts and the low thermal neutron cross-section of the Li7, Be and F atoms used in the salts makes them versatile for a variety of reactors.

The fluoride salts (mixtures of LiF, BeF_2 and fissionable and fertile fluorides) are excellent heat transfer agents. The volumetric heat capacity is high, tending to yield compact heat transfer systems. The high reactor mean temperature (1150°F) allows greater temperature differences in heat exchangers and thus smaller heat exchangers.

The low pressure of the liquid fuel reduces heat exchanger header thickness, pipe wall thickness, and reactor vessel wall thickness over the aqueous reactor. These factors tend to reduce the difficulty of maintenance operations. The low pressure also reduces the volume required within the secondary containment vessel, and should simplify plant construction.

The high melting point of the fuel is an advantage from the point of view that in case of a spill of fuel it will tend to solidify into a relatively water insoluble solid, thus tending to prevent spread of fission product contamination.

e. Limitations of the Molten Salt Concept

The high melting point of the fuel salt requires equipment for preheating the system to the reactor temperature. It also requires control of the coolant temperature so that it does not freeze the fuel in the heat exchanger.

Sodium reacts with the fuel to precipitate Uranium. If sodium is used as a primary coolant, the fuel salt must be pressurized with respect to the sodium so that any fuel leaks will be in that direction.

Oxygen in excess of limited amounts will react with the fuel to precipitate uranium, so that the fuel system must be buffered to prevent contamination by air or moisture.

The maximum conversion ratio and minimum doubling time are dictated by those attainable in a graphite moderated system. These are not as good as those obtained in the aqueous homogeneous reactor.

The alloy used for construction and the high purity fuel salts with their Li^{-7} content are expensive materials.

2. The Development of Low-cost Power Potentialities

a. The Reference Design for the Economic Plant

(1) Description

The reference design reactor* is essentially a cylindrical graphite assembly 12.25 feet in diameter by 12.25 feet high containing molten salt fuel in vertical channels constituting sixteen percent of the volume of the reactors. The fuel salt is circulated from the reactor vessel to four primary heat exchangers by four fuel pumps. An inert coolant salt is used as the intermediate heat exchange fluid, which is used to superheat and reheat steam in a Loeffler boiler system. The reactor develops 760 MW of heat. By using 2000 psi steam at 1000°F with 1000°F reheat, the net electrical output is 318 MW electrical.

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* For detail see ORNL-CF-59-1-26

(a) Proposed Fuel System and Cycle

The fuel salt has an initial composition of 0.3 mole percent UF_4, 13 mole percent ThF_4, 16 mole percent BeF_2 and 70.7 mole percent LiF. There are 900 cubic feet of fuel salt in the system, and the initial inventories are 829 kg of Li^7. The fuel salt is withdrawn from the reactor primary circuit at a rate equivalent to 20 percent of the fuel inventory per year and placed in 150 day hold-up tanks to await chemical processing. As the fuel is withdrawn it is replaced with fresh fuel of the original composition except that the UF_4 content is adjusted to keep the reactor critical. The uranium additions would comprise all of the mixed U-233 and U-235 recovered from the chemical plant together with fresh highly enriched U-235.

The equilibrium inventories of uranium isotopes in the reactor primary system are 510 kg of U-233 and 430 kg of U-235. The average conversion ratio is 0.67 so that 92.5 kg of U-235 per year must be added to the system. Thorium is burned at a rate of 188 kg per year.

Fuels

The above fuel solution would operate at a reactor inlet temperature of 1075°F and an outlet temperature of 1225°F (580°C and 663°C). The liquidus temperature for this system is 524°C, the first solid precipitating being 3 LiF • ThF_4. If the composition were changed by addition or removal of one constituent at 500°C, the following are the limits outside of which precipitation would occur: ThF_4, 8-21 mole %; BeF_2, 11-40 mole %; LiF 53-74 mole %.

No complications occur from heating. The solubilities of alkali fluorides and alkaline earth fluorides are quite high; trivalent fluorides, like those of plutonium, uranium and the rare earths will coprecipitate when the total concentration exceeds a critical limit. The solubility of rare earth fluorides has not been measured in this particular salt composition; from measurements on the similar system 70 LiF - 10 BeF_2: 20 UF_4 the solubility at 565°C is estimated to be 1.0 percent, or sufficient to accommodate the 0.4 mole percent of rare earth fission products that would build up during a nine-year fuel life. Certain heavy metal fission products, such as Mo, Ru and Nb perhaps form as elemental metals on the walls of the system. Heat or other effects from these might be a considerable problem. The total quantities of the three formed in nine years operation is estimated to be about 430 kg. If uniformly distributed, with a density of 7 g/cc, a layer of about 1 mil thickness would form.

An alternate fuel consists of 20 mole % UF_4 (1.3% enrichment), 10 mole % BeF_2, 70 mole % LiF. The liquidus temperature for this system is 500°C, the first solid precipitating being 7 LiF•$^6UF_4$.

If the composition were changed by addition or removal of one constituent at 580°C, the following are the limits outside of which precipitation would occur: UF_4, 12-28 mole %; BeF_2, 3-30 mole %; LiF, 61-76 mole %. PuF_3 is formed in this fuel as well as the rare earth fluorides. For pure PuF_3 in this fuel, the solubility limit is 1.38 mole % at 565°C. Thus precipitation appears to limit the unprocessed operating period to about 15 years.

Fuel Problems

Although many aspects of the molten salt fuel system appear promising, there are a number of uncertainties which might have a serious effect on the cost or operating procedure. These are as follows:

Possible reaction of gases in the graphite with UF_4, with formation of UO_2F_2 and UO_2, during initial charging of the reactor core with molten salt. Such a reaction will occur unless the graphite is completely free of air, moisture, etc. The graphite must be pretreated to remove all residual oxygen. Although no method of pretreatment of the graphite has been demonstrated, the project people are optimistic that treatment with HF or pretreatment with a spare charge of salt will reduce the oxygen to suitable levels.

Accidental air leakage. As described in the previous paragraph, UO_2 would be precipitated if air (containing moisture) leaks into the fuel system. Some UO_2F_2 and/or UF_6 might also be formed with resultant corrosion problems. If UO_2 is precipitated, processing of the fuel out of pile with H_2 and HF may be required which would be difficult on account of high radioactivity. This might require as long as two months, since it would not be practical to install sufficient capacity for doing the job in less time.

Burnable poison vs control rods. With the U-238 - U-235 fuel cycle, there is a significant increase in reactivity as Pu^{239} starts to build up. Poison must be added either as a burnable poison that is soluble in the fuel, or as control rods. For the ultimate reactor design, a preference has been stated for the former. No suitable poison has yet been demonstrated.

Change in oxidation states in fuel due to fission. When each U or Pu atom fissions, two nuclide species are formed, each of which (except for the rare gases) requires fluorine atoms to stay in solution. (If a sufficient number of fp left the fuel solution as nonfluorides, it is conceivable that an excess of fluorine atoms would be present, with the result that volatile and corrosive UF_6 would be formed.) At the end of 9 years continuous operation without reprocessing, about 0.6 m/o of the total metal atoms in the fuel solution will

have fissioned. Discounting the iodine and bromine formed there is a nearly exact fluorine balance if 1) the rare gases (22% of total fp) are removed as elemental species, 2) the rare earths (26% of total fp) form stable trifluorides, 3) Ru, Mo and Nb (22% of total fp) precipitate as metals and 4) other fission products (30%, 1/2 of which is Zr) are in solution as tetrafluorides.

One limit to continued burn-up without processing is the co-precipitation of the trifluorides. An earlier limit, however, may be the precipitation of Ru, Mo and Nb which may cause plugging of heat exchanger tubes. If such precipitation occurs (and this is the current view) it might limit the life of the heat exchangers with or without reprocessing, since reprocessing would not remove the precipitated metals. Some system for removal of these metals should be developed.

Materials

The basic material being considered as a container for the molten salt fuel system is a nickel-base alloy designed as INOR-8 (Hastelloy N, Inconel 806). It is a solid solution type alloy not subject to age-hardening. The only metallurgical instability is some carbide precipitation at long exposures to high temperature with some increase in strength but with no reduction in ductility.

Corrosion and Mass Transfer. This alloy appears to have no basic problem from a corrosion and mass transfer standpoint. No measurable attack has occurred during one year exposure in low uranium salt at 677°C (less than $\frac{1}{2}$ mil). At 899°C about one mil deep pitting attack was observed in 42 days exposure.

Irradiation, per se, is not expected to have any significant effect on these properties since the ability of INOR-8 to resist corrosion and mass transfer does not depend on "surface films". The corrosion and mass transfer that does occur is a consequence of the salt leaching out chromium from the surface of the container in the hot leg ($Cr^{\circ} + UF_4 \rightarrow CrF_2 + UF_3$), thereby decreasing the Cr content of the alloy until an equilibrium concentration of about 500 ppm is attained in the salt. In the cold leg, Cr is deposited on the metal surfaces; it there diffuses inward and increases the surface Cr content. The process will continue until steady state rate is set up, the lower solubility of CrF_2 in the cold leg being compensated by the higher Cr activity in the metal.

Weldability and fabricability of INOR-8 is comparable to Inconel and the standard Hastelloys. Inert gas shielded tungsten arc welding is used. No special difficulty has been encountered. Sufficient quantities have been produced to indicate that no problem will be encountered with respect to procurement.

Graphite. Penetration of graphite by molten salts in the LiF, BeF_2, ThF_4 system has not been successfully accomplished. These tests were run, however, with either poorly outgassed or as-received graphite. Unless further tests, including long time irradiation, show otherwise, it must be assumed that penetration will occur. Since the ratio of graphite to fuel volume in the core is much higher for the molten salt reactor than for the LMFR, a greater fraction of the fuel would be tied up in the graphite.

Other Materials. For valve seat application, a combination of Mo against a Co bonded TiC cermet has been satisfactorily used as a hard contact seat, and copper against molybdenum appears satisfactory as a soft-seating combination.

(b) The Reactor

This vessel must completely and indefinitely contain a high temperature fluid (1250°F), which also has a relatively high melting point (about 930°F), at a low pressure (about 100 psig). The vessel provides support for the graphite moderator and the means by which the molten, homogeneous salt mixture is admitted and exited from the critical region.

The reference design suggests a right cylinder about 12 feet in diameter and about 12 feet in height. This INOR-8 vessel is about $1\frac{1}{2}$ inches thick. A multiple loop pipe header is welded to the vessel at both poles providing the entrance and exit for the fluid. The molten salt flows upward through vertical channels constituting 16 per cent of the volume of the graphite moderator. About one-half of the salt leaving the critical region flows upward to an expansion tank, or dome, wherein some of the gaseous fission products are removed.

A relatively close fitting, low chrome steel vessel completely surrounds the reactor vessel. Helium gas, pumped through the annulus between the outside of the reactor vessel and the inside of the containment vessel, is used to cool or heat the reactor vessel and its internals. The helium gas pressure required is about 75 psig; therefore, a relatively thick containment vessel need not be designed. The support for the gas jacket probably needs to be integrated with that for the reactor vessel but present design practices can be used and no further development appears to be needed.

The use of the graphite moderator within the reactor vessel appears to present several design problems. It is desirable to try to minimize the variation in temperature of the graphite in a given horizontal plane during the period following drainage of the fuel from the reactor. The heat generation rate at this time depends upon the amount of fuel and fission products left on the surface of the graphite and the quantities that have penetrated into the graphite. Experimental

investigation of this problem, especially in in-pile loops, is necessary. It is envisioned that a considerable experimental effort is dictated with a total expenditure of the order of two million dollars. It has been shown that the thermal conductivity of graphite decreases with total exposure to fast neutrons; this effect is not known in the 1100-1200°F temperature range and should be investigated experimentally. A compromise between the most desirable nuclear configuration and the predicted heat transfer performance is dictated in the selection of the hole sizes in the graphite and their spacing. A reactor vessel mockup is required to insure that the desired flow distribution provides adequate cooling for the reactor vessel as well as the graphite moderator. It has been found that the fuel salt does not always wet the graphite and consequently the heat transfer work that has been done on salt systems does not apply; a very modest effort is required to determine the heat transfer between the fuel salt and the graphite.

Heat transfer between the metal walls and flowing fluoride salt mixtures has been defined by experimental work. Therefore, the designer is able to specify the flow rate required to adequately cool the vessel without further experimental work.

Very few nuclear instruments are required for successful operation of this reactor. It is not necessary for the sensing elements of these instruments to reside within the reactor vessel or within the containment vessel. It will be necessary to supply inert gases to the sensing elements and to adequately gas-cool them. However, there is no new technology involved.

Field fabrication of reactor vessels of the order of 20 feet in diameter and 20 feet in height is feasible with present construction practices. The wall thickness required for satisfactory performance of such a vessel is about four inches and this thickness is near the maximum for present field fabrication practices.

(c) Primary Heat Exchange System

The 318 MWE net reference design proposes the use of four parallel primary loops. The combined core and blanket salt flow flows from the reactor at an average bulk temperature of 1225°F through 18" piping directly to four primary loop canned rotor sump type circulating pumps each rated 8,900 gpm at 75 psi. The fuel salt then circulates through the tube side of four 5830 square foot primary heat exchangers where it gives up 195 MW thermal and returns to the reactor at 1075°F.

The loops and all principal equipment are fabricated from INOR-8.

The secondary fluid is a 65 mol percent Li^7F - 35 mol percent BeF_2 barren salt selected for compatibility with the fuel salt which in turn generates and reheats the steam used in the electrical generating facilities are presented as a Loeffler type boiler. If this feature of the plant should prove undesirable, a low-melting lithium, rubidium chloride eutectic salt could be used as a secondary fluid that is compatible with the fuel. The corrosion resistance of the INOR-8 alloy to the chloride salt has not been tested.

Heat Exchangers

No detailed reference design has been presented. Extensive work has been performed on the heat transfer characteristics of the fluoride salts and the design parameters have been optimized. The general description of the reference design is a bayonet type tube bundle mounted in a vertical shell to permit "semi-direct" replacement.

Although simple in concept the molten salt exchanger will be complex in detail because of the need for preheating the primary and secondary loops, provision for remote maintenance, drainability, etc. These, however, are problems of engineering detail only and seem to pose no insurmountable obstacle, although it may be anticipated that extensive design and development work will be required to produce a satisfactory configuration. The ultimate exhanger design should prove quite costly, running in the neighborhood of $100 per square foot of active surface.

The probability and consequences of leakage are not conditioning on this design since it is possible to maintain the secondary fluid at sufficient over-pressure to assure leakage of the secondary fluid into the primary fluid at reasonable rates with no deleterious efforts. Leakage of major proportions will require shutdown for maintenance and/or replacement of the affected component.

The low vapor pressure of the primary fluid permits the use of thin wall tubes and presumably not too heavy tube sheets. The benefits accruing from this consideration cannot be evaluated until a detailed reference design is established.

Corrosion of the exchanger and fuel hide-out do not appear to be serious problems on the basis of work to date. Further, the high thermal capacity of the fluid permits the use of a relatively small temperature differential across the reactor. This should further minimize the design problems associated with the transient thermal stresses of the tube sheets.

Fuel Circulating Pumps

No molten salt pumps with the size, performance and gas leakage requirements of the MSR concept have been built. However, molten

salt pumps up to 800 gpm have been operated successfully on non-radioactive loops and pumps rated up to 1500 gpm at heads to 350 feet have operated successfully at temperatures to 1525°F. These are vertical shaft centrifugal sump type pumps with the impeller on the end of the shaft.

Pumps tested to date have been of the oil-lubricated type with elastomer seals. For the long life requirements of pumps for the power reactor concept it is necessary to develop salt or gas-lubricated bearings, and low leakage rotating mechanical gas seals. Stationary elastomer seals could be replaced with metallic O rings.

Neutron and gamma shielding is required for protection of the motor windings and lubricant. The reference design indicates a gas-cooled shield of rather complex design with a lower salt-lubricated journal bearing, a mechanical gas seal, oil-lubricated motor bearings. Pumps of similar mechanical design have been used for molten salts in connection with the ANP program and it is believed possible to extrapolate this experience to the capacities required for a utility size reactor plant.

Piping

Although to date no large fittings or large diameter tubing have been procured, plate is available in INOR-8. It is believed that piping procurement will not pose a problem.

The principal difficulties associated with the piping are the design of loops with sufficient flexibility to reduce thermal expansion stresses to permissible levels. The thermal expansion of INOR-8 is of the same magnitude as the feritic steels. The low vapor pressure of the primary and secondary fluids permits the use of thin wall piping which is of some benefit in designing for mechanical flexibility.

The close coupled system proposed in the reference design would require flexible mounting of the reactor to obtain permissible thermal stresses in the piping between the reactor and the pumps. A detailed thermal stress analysis of the piping has not been made and it is doubtful whether the configurations presented would stand up to such scrutiny. The proposed primary system holdup volume may be unduly optimistic because of this factor.

Primary Auxiliary Systems

The functions performed by the equipment included in the MSR primary auxiliary system are: (1) heat addition and removal required during abnormal operations; (2) removal of volatile fission

products; (3) sampling, enriching, and removal of primary fluid; and (4) sub-critical storage of primary fluid.

All of the primary system components are heated to about 1200°F prior to charging of the fluid fuel because the proposed mixture of fluoride salts melts at about 950°F. The components are heated by circulating hot helium gas through the annulus formed by the primary barrier and the gas jacket. The heating system is compartmentalized to allow rough control of the rate at which the temperatures are changed. Some of the equipment, e.g. the drain tanks, will be maintained near operating temperature, although the equipment is not in use. The size of the blowers and gas or oil fired or electric heaters is determined by the particular components to be heated by them. It is expected that the sum of the heating capacities of the furnaces will be about 2 MW and the total blower horsepower will be about 1000 HP, assuming an acceptable temperature rise of the helium of 200°F through the furnace and an operating pressure of about 75 psig. The reactor vessel heating system is interconnected to a gas-to-water heat exchanger capable of removing about 3 MW. All of this equipment is located in a semi-radioactive area because it may become contaminated during operation. The floor area required to house these auxiliaries is about 7500 square feet.

The gaseous fission products are collected in the expansion tank which is attached to the top of the reactor vessel and also in the sumps of the primary pumps. The pressure in the gas spaces (volume of about 250 ft^3) is allowed to increase until about 5 psig overpressure is evident (occurs in about one month) and is then carefully vented to a hold tank of about 1000 ft^3 volume. The gases are held in the large tank for six months to a year and then compressed into high pressure cylinders; it is anticipated that these cylinders will be buried. The solubilities of the relevant gases in salts similar to the proposed fuel salt have been determined but their solubilities in the particular fuel salt chosen have not been examined experimentally. In addition, it appears that the behavior (e.g. fugacity) of the fission product halogens in the fuel fluid should be determined experimentally. A modest expenditure is indicated (approximately $200,000).

Three batchwise manipulations are required to sustain steady state operation of the MSR. Daily additions of fuel are required to reconstitute the fuel, and daily samples by means of a mechanical lock and vertical transport assembly which handles solid cylinders of fuel salt mixture. The assembly penetrates one primary pump sump. This assembly has not been designed but it does not appear that any development problems are involved. Fluid fuel may be removed from the primary system by controlled, pressurized transfer from one pump sump to a spent fuel storage tank; a right cylinder five feet in

diameter is capable of storing the spent fuel fluid. This vessel need be heated only during transfer from it to the chemical reprocessing plant or to shipping casks; it is jacketed and is heated or cooled by means of circulating helium gas.

It is necessary to drain the fuel fluid from the reactor and primary heat exchange loop prior to some maintenance operations. During the course of these operations the fuel fluid is stored in an array of inclined pipes contained within a shell; the unit consists essentially of a cross-flow heat exchanger with cool gas entering at the top and exiting at the bottom. About 2700 square feet of active surface is required in each exchanger. This method of temperature control is utilized to eliminate the need for remote maintenance of electrical resistance heaters, bayonet tubes and the like. The helium flow rate needed to maintain the salt temperature at 1600°F is about 3000 pounds per minute. Five blowers of about 250 HP each (1 standby) appear to be required. These blowers are also those used for the heatup cycle. Two atmospheric water boilers of moderate size are suggested to remove the heat from the recirculating helium gas; the heat transfer area of each is about 500 ft^2 and the water rate to each is about 100 gpm. All of these components can be designed from present technology; however, it does appear prudent to determine the afterheat generation.

Fuel fluid is transferred to the drain tanks from the reactor system by pressurizing the reactor system with clean helium gas. Fluid fuel is pumped into the reactor system from the drain tanks by means of a vertical sump type centrifugal pump set in a closed sump located at the low point of the drain tanks. The capacity of the pump is about 200 gpm at a discharge pressure of about 25 feet of fluid. During the transfer the reactor vessel is vented to the hold-up tank of the off-gas system; the volatile fission products in this gas are removed and the partly contaminated helium gas is compressed and stored in gas cylinders for re-use.

(d) Secondary Systems and Steam Generators

The secondary system consists of an intermediate coolant salt transferring heat from the primary heat exchangers to steam superheaters and reheaters. A large portion of the superheated steam is used to generate steam in a Loeffler boiler. A steam blower circulates steam from the boilers to the superheaters. Problems of the Loeffler system are discussed in Section (f) below, under Maintenance.

The Loeffler boiler system imposes a capital cost penalty to to the molten salt system. It was chosen so that a high melting point fluoride salt that is compatible with the fuel could be employed as a single heat transfer loop between the fuel and steam systems.

(e) Control, Instrumentation and Operating Characteristics

The feasibility of the operation of a molten salt fueled reactor at high temperatures was demonstrated in 1954 in experiments with the Aircraft Reactor Experiment (ARE). The reactor was equipped with a conventional control rod system, but it was demonstrated that control could be achieved by use of the negative temperature coefficient of reactivity. It was found that the reactor would follow any changes in the heat extraction system.

The reactor proposed in the reference design would make use of the negative temperature coefficient for its primary control. It may be necessary to have control rods, as in the case of the LMFR, to limit excessive temperature rises in the core due to a temporary excess of fuel. Only preliminary work has been done on the possible start-up, shut-down, and system operational procedures. The start-up operations of this reactor system will be complex due to the need of preheating all equipment and piping; likewise, in shut-down operations, this equipment will have to be cooled. Means have to be provided for preheating and cooling of both the primary and secondary systems. The time required in start-up will be determined to a great extent by the time required to preheat the systems. The start-up of this plant requires either steam from an outside source or electric heaters to start the Loeffler cycle.

Once the reactor is critical and the secondary system and the Loeffler cycle are in operation, control would be a combination of negative temperature coefficient, variation of fluid flow in the secondary system, and variation of steam flow in the Loeffler cycle. Information on plant stability and its kinetic behavior for disturbances in flow, power, and reactivity will have to be obtained from analogue studies and reactor experiments. This information would aid in designing an over-all plant control system.

Extensive instrumentation will be needed for start-up and shut-down operations particularly with regard to thermocouples. These will be required to indicate the thermal conditions of all piping and equipment. An additional problem during start-up and shut-down, when there is no flowing molten salt in either primary or secondary systems, is the possible distortion of the pump internals due to uneven heating or cooling. This distortion could prevent the start-up of the pumps.

In addition to operation of the main plant, there is also the operation of supporting systems. These supporting systems will need extensive control circuits and instrumentation to indicate that they are functioning properly. The supporting systems would

include the dump tanks, cell cooling equipment, off-gas system, and equipment for maintaining an inert gas atmosphere in the reactor cell.

An area, which is important in the control of the reactor, is the control of fuel inventory in the system. This will require some mechanism which would obtain a sample of the molten salt, the sample would be moved to a hot cell where it would be analyzed for its uranium and thorium content. The control of the inventory of molten salt in the primary system may require that the dump tanks also act as weigh tanks, particularly when filling or dumping the primary system. An alternative for weigh tanks would be a system of tank level indicators.

The proposed reference design is still in its very early stages and a great deal of work remains to be done before a detailed operating procedure can be developed.

(f) Special Maintenance - Facilities and Problems

Inherent Problems

Many of the problems in maintenance associated with the molten salt reactor concept are similar to those encountered with the molten bismuth reactor concept. The fuel carrier (LiF, BeF_2) has a very much higher melting point than the ambient temperature, therefore preheating of the plant equipment prior to the introduction of the molten salt is a necessity. Likewise, a cooling system is needed to cool the reactor after the molten salt is removed from the system. This concept has an additional problem in that the secondary system uses a fluoride salt, which has a high melting point. The necessity of this fluoride salt is for reasons of compatibility with the fuel in case of a leak between the primary and secondary system. The secondary system would therefore have to be equipped with a heating system.

The proposed heating and cooling system for the reactor vessel, primary pumps and heat exchangers, would use helium gas. This necessitates a gas-tight outer container around each piece of equipment and piping. This type of construction greatly complicates the problems of maintenance, with the greatest difficulty arising when piping in either the primary or secondary system has to be cut to remove a major component. No methods have been proposed for carrying out this type of operation but methods developed in the molten bismuth concept would be applicable for this concept. It has also been indicated in the design that some parts of the primary system would be heated by the use of "conventional electric heater-insulation"; the short service life of this type of equipment may cause extreme problems in maintaining such a system remotely.

A fuel fill and drain system has been provided to serve as a molten salt storage facility before the plant is started and as a drain system when the primary system has to be emptied. The maintenance of the drain tanks may prove very difficult due to the number of connections that have to be broken remotely for removing sections of it. An additional problem will be the maintaining of the extensive instrumentation that will be needed.

The reactor vessel with its graphite core is the largest component in the plant. No maintenance except on control rods would be performed on the reactor proper, e.g. vessel, graphite, and any repair work, replacement of graphite or work on the reactor vessel is considered a plant modification.

Only very preliminary design work has been done for the primary pumps and heat exchanger. For the primary pump, the final design would be one in which the entire pump is removed and replaced as a unit, with the pump casing as a permanent part of the piping system. For the primary heat exchangers, the design would be of the sump type so only the tube bundle would have to be removed.

The auxiliary systems for the molten salt fuel reactor include molten salt transfer equipment, which includes the dump tanks, off-gas system, and preheating and cooling system. The fuel transfer system uses a vertical sump type centrifugal pump, together with its associated valves which would be remotely maintained. The problems associated with valves for molten salt fuels are those of maintenance of allignment and self-welding of the closure. The off-gas system is used for the continuous removal of fission product gases. This system, because of the high activity, would have to be maintained remotely. The proposed gas preheating and cooling system would use a package type maintenance, that is the blowers, heaters, and cooling coils would be in some type of container which could be removed remotely after all connections were broken. No detailed work has been done on this container.

A ventilation system is used to prevent excess temperatures in the reactor cell. This cooling of the cells is done by means of forced gas circulation through radiation type space coolers. A cooling medium such as Dowtherm, in a closed loop removes heat from the space coolers and dumps it to a water heat exchanger. The gas blower and cooler would be built so that it could be removed and replaced as a unit.

The secondary system utilizes a fluoride salt as the heat transfer medium. For this system maintenance will be completely contact after the system is drained. This secondary system will operate at a higher pressure so any leakage would be into the primary system.

Although the feasibility of the molten salt concept does not depend on the use of the Loeffler steam cycle, some consideration will have to be given to the maintenance of this system. All maintenance on the system would be direct contact as there are no problems of radioactive equipment. Two important items in the Loeffler cycle which may cause major maintenance problems are the steam blowers and the steam nozzles in the boiler section. Much development work is needed to obtain a large dependable steam blower. The solution to the problem of erosion of the steam nozzles may require an extensive research program.

The instrumentation and methods of maintenance of these various instruments has not been gone into in the reference design. A leak detection system will be needed for the primary system since all components and piping are enclosed by outer sealed containers. The detection system must serve two functions; show that a leak has occurred and, give the location of the leak.

Methods Employed to Solve Maintenance Problems

The maintenance concept employed in this design would be called "dry semi-contact", that is, using both contact and remote methods for the primary system. The layout of the plant consists of rectangular canyons with the equipment located below the floor of the canyon. Located in the canyon floor are removable plugs over major equipment. The reactor, primary pumps and heat exchangers are located in a cell which is steel lined and is gas-tight. An inert atmosphere is maintained in the cell at all times. The canyon above the biological shield, which forms the roof of the reactor cell, is sealed and provided with a crane, boom mounted manipulators and viewing windows. This room would be accessible to personnel to allow carrying out certain maintenance operations.

In the operation of replacing a failed primary pump, after the primary system is drained, service connections such as power leads, instrument leads, gas lines, and water cooling lines would be removed manually. All personnel would leave the compartment and the removal of the pump into a container would be carried out remotely with the aid of the viewing windows. The pump is transferred to a storage coffin, and a new pump is lowered into place. After the compartment is decontaminated, personnel will again enter and make up all service connections. The replacement of a heat exchanger would follow much the same procedure with the exception that the secondary system pipes and a seal weld on the head of the heat exchanger would have to be cut manually. A problem which arises during the changing of primary pumps and heat exchangers is the need of preventing large amounts of air from entering the primary system.

becomes critical, at which point the temperature can be lowered no further because of power generation. Finally, UF_4 will be added to bring the reactor to its design operating temperature, as a trimming operation.

As long as no repairs are necessary on the primary fuel loops, the reactor will be on continuous operation, with both fuel and coolant salt pumps going. At zero power operation, the reactor fuel circuit and coolant salt loop operate isothermally at 1150°F, the mean critical temperature. When there is a demand for power, steam is started into the superheater and the reactor picks up the load and will automatically adjust to any load. There is no rate of loss or pick-up of load that can cause nuclear or temperature overshoot difficulties in the reactor*. Load change rates will be limited by allowable thermal strains caused by rapid temperature changes in heat exchangers and piping.

When repairs are necessary on the primary loop, the fuel will be drained and the loop cooled. On restarting, the fuel and the reactor should be heated above the critical temperature before the fuel is brought into the reactor circuit, since it already contains the critical concentration of uranium. Simulator studies would indicate that no hazard is involved even if this precaution is not properly followed.

Control of the reactor is primarily by thermocouples to indicate temperatures. Although thermocouples will be located throughout the system, it is intended that routine reactor control (achieved by UF_4 or ThF_4 additions) would be based on thermocouples in the coolant salt circuit where they are readily available for maintenance. Other critical instrument requirements are those to tell whether the pumps are running, and liquid level indicators to determine the amount of fuel and coolant salt present.

Hazards

The following incidents may be considered: A steam leak into a coolant salt circuit would lead to drainage of that coolant salt loop and repairs to the superheater. The reactor need not be drained nor need the fuel pumps be stopped. Afterheat could be dissipated through the undamaged superheaters.

Temporary loss of all coolant salt pumps would not cause a sudden temperature overshoot in the reactor. If circulation of at least one of the coolant loops can be established within 15 minutes, draining of the reactor would not be necessary, since the temperature rise due to afterheat is not catastrophic.

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* See ORNL-2634, p. 46

The proposed fuel solution has limited oxygen tolerances to prevent the possible precipitation of UO_2 from the molten salt fuel. After a replacement of a pump or heat exchanger it will be necessary to purge the primary system before the molten salt fuel is introduced.

The design of the molten salt reactor is not far enough along to have considered in detail the performance of maintenance below the main biological shield in the reactor compartment. Provisions will have to be made in the design to be able to replace the primary heat exchanger shells, pump casings, primary piping and all instrumentation not accessible from the access canyon. This would involve the use of remotely controlled equipment, perhaps similar to that proposed for the LMFR and PAR.

Maintenance Equipment

There has been no extensive development program of remotely operated tools for the molten salt reactor. Work has been carried out in the use of boom mounted manipulators for the disassembling and assembling of small pumps. However, the use of this technique in a large reactor plant would not be practicable except in a hot maintenance shop. Development work has been carried out on designing a flanged joint for small pipes. This is a freeze flange joint that consists of a conventional flanged ring joint with a cooled annulus between the ring and the process fluid. The salt that enters the annulus freezes and provides the primary seal. The ring provides a back-up seal against salt and gas leakage. The use of a flange connection may complicate the preheating and cooling system and its use might be a disadvantage.

The maintenance equipment being developed for use with the LMFR and PAR concepts would be applicable to the molten salt concept with changes in plant layout.

(g) Major Spare Equipment Requirements

Major spare parts carried on hand would include one each of the removable cartridge portions of the pumps and primary heat exchangers and one each of the superheaters and reheaters.

(2) Reactor Operating Characteristics

The initial start-up of the reactor will require preheating both fuel and coolant salt loops, and then filling these loops with hot salt from the drain tanks. Preheating may require several days. After filling, circulation of the fluids by the pumps will be started. After a check-out period, the reactor will be brought critical by additions of UF_4. When criticality is approached, it will be done by gradually lowering the temperature of the fuel until it

Loss of one fuel pump would not have any temperature overshoot consequences, but would be cause for an orderly shut-down of the plant, draining the fuel and replacement of the pump.

If either a fuel pump or a coolant salt pump should fail, the steam supply to the corresponding superheaters must be automatically shut off to prevent freezing of coolant salt in that circuit.

Loss of all fuel pumps at once would cause a temperature overshoot in the reactor due to the sudden gain of delayed neutrons in the reactor core. The exact magnitude of the overshoot has not been determined for this reactor but it would probably not exceed 200°F. The principal difficulties would result from thermal stresses (not yet calculated). Control or poison rods can control this overshoot. It would be preferable, however, to have auxiliary power available to start one of the fuel pumps.

The principal nuclear hazard anticipated would result from allowing sufficient oxygen into the system to precipitate UO_2. The UO_2 would tend to settle out in quiescent regions of the fuel circuit. Conceivably, a critical mass could accumulate, or a slug of held up UO_2 could be released and circulate into the core placing it on a fast period leading to a severe temperature overshoot. Measures needed to prevent these happenings are: great care in prevention of oxygen access to the fuel; design of the circuit so that UO_2 cannot settle out at any point other than at a designated one; inclusion in the circuit of a trap where settling can occur; frequent check on the chemical composition of the fuel and petrographic examinations for the UO_2 phase. If these measures should fail, the ultimate consequence would be a rupture of the primary circuit and a spill of fuel into the containment space.

The volume within a reactor containment structure is usually adjusted to accommodate the total gas generated at a reasonable pressure. Since no gas producing reactions have been discovered in the molten salt system, the reactor containment structure has no pressure requirements other than those needed for testing it and the volume is dictated by convenience. One proposal is to line the primary system cell with two layers of thin sheet steel. At critical places (penetrations) the space between would be used as a buffer zone so that an inert gas can be kept in the primary system cell.

Potentially Achievable Cost Reductions

During the analysis of the plant costs for the MSR it became apparent that the choice of the Loeffler boiler system was a poor one, adding substantially to the capital cost and requiring a

separate development program. A rough analysis on two different substitute systems indicates savings of about $4,200,000 in direct labor and material, or $7,300,000 after top charges. This charge would result in a saving of 0.31 Mills/Kwh in power plant investment costs and 0.07 Mills/Kwh in calculated maintenance costs, or a total saving of 0.38 Mills/Kwh.

INOR-8 prices have been taken from the bids submitted for the first commercial production of moderate sized orders of single lots of material. Low bidder quotations range from $2.65/lb for 0.25" sheet (20,000 lb lot from Haynes Stallits) to $15/lb for ½" x .045" wall seamless tubing (from Superior Tube). The average price assumed is $6 per lb. It may be expected that if a reasonable market develops for this material its average price might drop to $3/lb. This would effect a capital cost reduction of at least $3,500,000, equivalent to 0.21 Mills/Kwh.

In the opinion of the project director, the basis used for evaluating chemical processing costs almost entirely obscures the value of the fluid fuel reactor concepts in achieving low fuel cycle costs. It does this by assuming that each power station is an isolated 333 MW plant, and that this plant must stand alone as far as reprocessing the fuel is concerned. The approach used also eliminates any chance for evaluating differences in chemical processing costs that may exist between the three reactor concepts.

In the case of the molten salt reactor, the choice of a graphite moderated reactor for evaluation by the Task Force was originally made on the basis that it required less frequent chemical reprocessing that a homogeneous reactor without graphite to achieve a moderate conversion ratio, and that therefore the fuel cycle costs and total power cost would be lower. The insertion of an on site chemical processing plant casts doubt on whether the best molten salt plant was actually chosen for evaluation, since the omission of graphite from the system would eliminate one element of doubtful feasibility from the molten salt reactor system.

With regard to the necessity for an on site chemical processing plant for the graphite moderated reactor, it is granted that, despite the planned precautions, it is possible that the fuel could become contaminated by some accident. However, the proposed on site plant would provide only for the recovery of the uranium, throwing away the salt carrier, thorium and lithium-7, and would do this at a rate requiring five years to reprocess the entire charge. The same operation could be accomplished more quickly in a large off site plant, and at less cost. The savings would come in a reduction of the on site capital costs and in the resulting chemical plant operating costs.

The ORNL estimate of direct labor and materials involved in providing facilities for canning the fuel, storing it, and providing shipping casks for sending it off to a central processing facility is $1,230,000. This item would replace the $3,600,000 listed for the on site recovery process. The chemical laboratory would not need to be as extensive as when on site processing is required so that $250,000 is estimated for two cells replacing the $700,000 for six cells. The savings in these two items alone over the proposed on site chemical plant is $4,730,000 capital cost after indirect, overhead, and contingency factors are applied to direct costs. It is believed that there should also be substantial savings in both capital and operating expenses for the gas facility and waste facilities since these would be used only intermittently and need less capacity if off site reprocessing were used. Also it is not believed that the Xe and Kr bottling facility will be as expensive as listed in the Chemical Reprocessing section. A reasonable figure for the capital investment of the remaining on site facilities of a chemical nature might be $5,000,000, and a reasonable operating cost for the largely intermittent operation, $500,000 a year. Thus the on site capital and operating costs mighs be $1,200,000/year, or 0.54 Mills/Kwh instead of the $3,460,000/year or 1.56 Mills/Kwh used for power cost estimates.

With off site processing, the remainder of the costs associated with the fuel cycle are as follows, based on the nine-year reprocessing cycle submitted to the Task Force. The fuel cycle costs listed below, are comparable to the total of those usually listed for solid fuel element reactors.

	$/Year	Mills/Kwh(net)
Uranium inventory charges	615,000	0.28
Thorium and fuel salt depreciation	515,000	0.23
Burn-up of U-235 and thorium	1,436,000	0.65
Fuel recovery costs*	326,000	0.15
		1.31

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* Sinking fund to provide: Cans for shipment of fuel $150,000; Cost of shipping 1300 tons 1000 miles round trip at $500/ton; Cost of chemical processing and waste storage in plant described in IDO-14363 = $960,000; Inventory charges during hold-up, processing losses, and depreciation of U-235 price from $17 to $15/gm = $1,820,000.

The development of a continuous salt purification process to eliminate fission products as they occur would reduce the fuel cycle costs by 0.52 Mills/Kwh. From this must be subtracted any additional capital and operating costs incurred by the improved chemical processing scheme.

In the opinion of the project director, the $11,800,000 chemical processing facility should have a through-put capability several times that assumed. The reason for this belief is that the half cubic foot per day capacity is only two days per week operation of the ORNL volatility pilot plant. If the net capacity of the chemical plant were even twice that assumed, a second power plant built at the same site would avoid the chemical plant capital and operating charges, or, if spread over two plants, these charges would be reduced by a factor of two on each plant.

b. The Development Program (Project Director's Appraisal)

(1) Some of the important objectives of a development program aimed at an economic reactor are:

The development of reactor components that are reliable and easy to maintain in sizes suitable for a power reactor.

The design and development of reactor layouts that provide for low capital cost and inexpensive maintenance.

The determination of the behavior of fission products as they are produced in a reactor over long periods of time.

The development of the best graphite structure for use as a moderator in a molten salt reactor, and the determination of the degree of penetration of the graphite, how to control it and the effects thereof.

The development program proposed would involve two stages, centering around the design, development, construction and operation of two reactors. The first would be an experimental reactor of 30 Thermal MW capacity whose design and construction is estimated to cost $18,000,000. The research and development program associated with the experimental reactor is estimated to cover $19,000,000 over a four-year period.

Upon the successful completion of the experimental reactor program, a project for the design, development and construction of a prototype power reactor would be undertaken. Such a reactor would probably cost $50,000,000 and a $35,000,000 research and development program extending over five years is proposed to enable the successful construction of this reactor.

At the end of this nine-year program costing a total of $122,000,000, enough information should be available to enable the design and construction of a commercial plant and to estimate its cost with reasonable accuracy.

The accompanying table shows a possible problem breakdown of the research and development funds over the next five years.

(2) The principal development requirements for breeder reactors are the development of a low-cost continuous chemical reprocessing method and the development of a reliable graphite core vessel that reduces leakage of neutrons to a very low value.

Of the approaches available for chemical reprocessing, the most interesting involves three parts. One is a very fast stripping of noble gases from the fuel as they are formed. If they can be stripped on a 30-second cycle, about 34.3% of the total fission products can be removed in this way. The second part utilizes the fact that rare earth trifluorides from solid solution with each other, and co-precipitate from salt solutions. By adding CeF_3, which is not a serious poison, so that it holds a concentration of about 1 mole percent in the fuel, and stripping it out by a cold trap so that a complete change of CeF_3 is made once a month, the rare earth fission product poisons can be held at their 30-day value. This would remove another 26.6 percent of the fission products. Finally it is believed that the noble metals such as Mo and Ru will tend to plate out of the salt. The problem is that of finding a suitable way of making this happen in a harmless place. This would remove another 23.3 percent of the fission products. Low cross section zirconium accounts for 80 percent of the fission products remaining after these treatments, and there do not appear to be any high cross section residual elements.

The provision of this or some alternate low cost chemical processing scheme is vital to a breeder reactor and if breeding is set up as a serious goal, a million dollar-a-year program would be proposed as an adjunct to the experimental reactor as it comes into operation.

Table IX-1

First Five Years - Molten Salt Reactor Development Program

(Thousands of Dollars)

Subject	FY 1960	FY 1961	FY 1962	FY 1963	FY 1964
Fuel Studies					
Behavior of fission products	50	50	50	200	200
Methods of Fuel Analysis	--	150	200	150	100
Solubility studies	100	100	50	50	50
Volatility of constituents	50	50	--	--	--
Burnable poisons	--	--	50	--	--
Thermal & physical properties	100	100	100	100	100
Methods of chemical processing	200	300	300	400	500
Preparation of experimental batches	75	75	75	75	175
Metal Studies					
Metallurgical & mechanical properties	75	75	75	--	--
Remote Welding	75	100	100	100	100
Remote inspection of welds	50	100	100	50	--
Long term out-of-pile corrosion	350	300	200	200	200
In-pile corrosion tests	375	300	200	200	200
Graphite Studies					
De-oxidation of graphite	150	150	--	--	--
Absorption of fuel, incl. in-pile tests	300	375	475	200	--
Improvement of graphite imperviousness	50	100	100	100	--
*Development of graphite core vessel	--	--	--	200	600
Engineering Development					
Basic pump development	150	150	150	100	150
Pump for experiment	300	400	400	--	--
Pump for prototype	--	--	--	400	700
Heat exchanger for experiment	100	350	300	200	--
Heat exchanger for prototype	--	--	--	100	600

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* Items required primarily for the breeder.

Subject	FY 1960	FY 1961	FY 1962	FY 1963	FY 1964
Engineering Development (Cont'd)					
Remote maintenance methods	300	400	400	300	400
*Rapid off-gas removal system	--	--	150	300	150
*Rare earth fission product removal	--	150	250	500	300
Dump valves	--	100	100	--	175
Fuel sampling and enriching	100	100	100	--	100
Other components, including instruments	150	225	225	375	900
Basic chemical and metallurgical studies	300	300	300	300	300
Design Studies	450	300	300	300	500
Project Administration and reserve	200	250	250	250	350
Totals	4,000	5,000	5,000	5,150	6,850

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* Items required primarily for the breeder.

3. The Development of the Breeding Potentiality

a. Two-Region Homogeneous Breeder

The conceptual design of a two-region homogeneous reactor plant is described in ORNL-2634. It consists of an 8' diameter core surrounded by a 2' thick blanket. The core vessel is 5/16" thick and made from INOR-8.

The performance of this reactor when fueled with the salt 7 mole % ThF_4: 58/7 mole % LiF: 34.7 mole % BeF_2: 0.6 mole %. $U^{233}F_4$ is given by L. G. Alexander in ORNL-2626. It has an initial inventory of 1238 kg U-233, but the increase in inventory requirements exceeds the production of U-233 for the first ten years, so that it requires 1377 kg of U-233 to start the reactor and carry it through ten years. After the first ten years, the generation of excess U-233 holds its own with the increase in inventory for at least the next ten years.

With greater elapse of time (more than 100 years) the build-up of U-235 and U-236 will decrease the conversion ratio by 0.07. This results from the fact that this is an epithermal reactor (mean neutron energy for fission = 40 ev), and eta for U-235 is excessively low at these energies. Two expedients would raise the conversion ratio to 1.000 even in equilibrium with U-235 and U-236. One of these is a faster chemical reprocessing cycle than the once-per-year proposed by Alexander. The other is the use of a sandwich construction for the core vessel, using either a honeycomb construction for stiffness or a filler metal of low cross section. This expedient would reduce the thickness of high cross section INOR-8 in the core vessel.

This reactor has less inherent difficulties of construction than the reference design "economic" reactor, since graphite is not involved. Its capital cost at the same capacity would be about the same, since the reactor itself would be less expensive but there is added the cost of a blanket circulation system. The operation and maintenance would probably be the same. The fuel charges might also be about the same, since the savings in fuel burnup would probably be compensated for by the cost of the increased volume of reprocessing (about 2 cubic feet per day instead of 0.5 cubic feet per day). Thus, this breeder, once supplied with U-233, would probably produce power for about the same cost as the "economic" reactor.

The initial supply of U-233 could be manufactured in the blanket of this reactor using U-235 fuel in the core, as in case 31, page 176 of ORNL-2634. The manufacture of the initial inventory of U-233 would take 18 years of 80% load factor operation. During

this period, the fuel cycle costs would be 1.9 Mills/Kwh more than those for the "economic" reactor. 3270 kg of U-235 would be burned, representing the diffusion plant output from 675 tons of natural uranium. The equivalent burnup on the natural uranium involved would be 4700 MWD/ton. After this induction period 1377 kg of U-233 would be available for this or other breeder reactors, to produce power from thorium.

A reactor of this size and type could be risked after the successful operation of an experimental reactor. Thus design and construction could start in five years, operation in another four years, and the U-233 charge accumulated in 18 more years. It could then operate for 20 years on thorium without the improvements of better processing and laminated core wall and by this time (47 years hence) these improvements can be expected.

b. One-Region Graphite Moderated Breeder

The reference design "economic" reactor, with two modifications, could just barely breed. These modifications are (1) increasing the core size to 20' diameter x 20' high, and (2) development of the continuous salt purification scheme described above. An inventory of 1025 kg of U-233 is required. This reactor is not capable of producing its own initial inventory, although if started with U-235 it would eventually approach self-sustaining operation with only thorium feed.

The larger reactor would add about $5,000,000 to the plant costs (over the economic reactor). After U-235 additions become negligible, the fuel charges would be about 0.5 Mills/Kwh less than for the economic reactor. If the continuous processing plant could be built for $13,000,000 ($1,200,000 more than the present plant) this breeder or very high conversion reactor would about break even with the "economic" reactor after conversion of U-233 operation. It does, however, require development of the continuous salt purification scheme.

Starting with U-235, 10 years of operation at 315 MW (electrical) would suffice to reduce the U-235 feed to less than 5 percent of the fuel requirements. It would take an investment about 1500 kg of U-235 to get this reactor over to the thorium cycle.

Construction would be started on this reactor with the same timing as for the "economic" reactor, or in about 10 years if the program took the orderly steps of an experimental reactor followed by a prototype reactor.

c. Two-Region Graphite Moderated Breeder

A typical two-region graphite moderated breeder reactor has a graphite core 4.4' in diameter by 4.4' long. The core volume is 85% graphite, 15 percent fuel. The fuel salt is the same as that for the "economic" reactor, and the blanket salt is the same, but without uranium. A two-inch graphite wall separates the core and blanket. The 30" thick blanket is pure salt containing 13 mole percent ThF_4 and no graphite. The volume of fuel in the core is ten cubic feet and it is assumed that with 30 cubic feet of external volume, 100 thermal MW can be dissipated. The core salt is purified on a once-per-month cycle using the salt purification process described in the development section. The blanket salt has its uranium content removed monthly by the fluoride volatility process.

The points of doubt with regard to the technical feasibility of this reactor are the construction of a graphite core vessel and its connection to the external system, the achievement of an external fuel volume of 30 cubic feet that delivers 100 MW to a secondary system, and the development of the salt purification system.

The graphite core wall need not be completely impermeable to the salt, since salt of the same basic composition is used in both core and blanket. Leakage of traces of uranium to the blanket would not be harmful since the uranium is removed from the blanket and returned to the core once a month. As to the connection of the graphite core to an external system, B&W have successfully operated such a connection to a 6" diameter pipe. A suitable graphite core vessel has not yet been designed, however, and the development and testing of such a structure that eliminates leakage of neutrons from the blanket could require a million or more dollars of development effort.

A high performance external heat transfer system requires extremely close coupling and the use of small diameter heat exchanger tubes. Experience is available for the development of components for such a high performance system. Maintenance on such a closely coupled system would be difficult and might be confined to the pump.

The neutron balance and inventory of this reactor after 20 years of operation is nearly the equilibrium case and is therefore indicative of its long term performance. The indicated doubling time is 44 years. An initial inventory of U-233 could be accumulated from the blanket in 3-1/3 years of 80% load factor operation with U-235 in the core. Such operation would require about 100 kg of U-235.

Information is not available at present for estimating the cost of power from this reactor.

X. LIQUID METAL FUEL REACTOR

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X. LIQUID METAL FUEL REACTOR

A. Objectives

The objective of the LMFR program has always been the development of economical power. For this reason the primary aim has been to find a system which would use relatively cheap, commercially available coolant and container materials and would give good thermal efficiency, high utilization of fuel and the possibility of integrated fuel processing. The development program has been given limited support consistent with the fact that the concept was a very advanced one.

Successive evaluations have gone deeply into plant arrangement, auxiliary systems, remote maintenance, fuel system stability, hazards, etc. While the capital investment has increased to some extent as more detailed consideration has been given, no unanswerable problems have been uncovered. Therefore, as a result of the continued favorable findings of the research and development program and the continued indication that this concept has application as a large scale nuclear power plant, the objective of the LMFR program is to build and to operate a small experimental reactor demonstrating the concept, followed by a prototype experiment to provide the engineering knowhow for construction of a large scale nuclear power plant on a commercial basis.

1. Low Cost Power

The earliest commercial plant would be of the one-region converter as described below. The fuel would be a slurry of ThO_2-$U^{235}O_2$ in bismuth. As the U-235 burned out and was partially replaced by bred U-233, more concentrated ThO_2-$U^{235}O_2$ slurry would be added to maintain criticality. The fuel would be processed on a very slow cycle; volatile fission products would be removed in the off gas system.

Such a plant of 333,000 KWe (gross), 825 thermal MW, could be built in 1969-1973. The estimated unit costs based on gross output would be:

	Mills/Kwh Gross
Power Plant Investment ($286/KWE)	5.72
Chemical Processing Plant Investment ($35/KWE) *	.71 *
Fuel Inventory, Use Charges and Burnup	1.36
Chemical Processing: Operation & Maintenance	.75 *
Power Plant Operation & Maintenance	1.46
Total Power Cost - Gross	10.0
Total Power Cost - Net	10.7

In the opinion of the Project Director, chemical processing costs would be lower than those shown here. *

Cost of power in Mills/Kwh would be expected to decrease after further development work. For example, more experience on materials and corrosion inhibitors in the fuel may allow operation at a temperature rise across the reactor greater than 300°F with a resulting increase in power. Although past research has been directed at using low cost steels, more expensive clad materials (e.g. Mo or Ta) may more than pay for themselves in higher temperature, Δ T and power, in the reduction of cost of power.

Operating expereince will permit improvement in design of systems and components and reduction in safety factors, which for the first plant will be high, based on very limited information. Various areas for cost reduction could be: elimination of the intermediate system, elimination of control rods, increase in steam temperature, reductions in the cost of graphite, simplification of both heat exchanger and pump designs which are conservatively designed at the present time. These and other changes will result not only in capital cost reductions but as operating experience is gained and development results improve component reliability, maintenance costs should be reduced.

Increased capacity is relatively easy to obtain in LMFR by adding heat exchange equipment since the reactor itself does not limit power. Increases would be limited only by the capacity desired in a single unit. As the size gets larger, on site chemical processing also becomes more economical. This would lead to better neutron economy and lower fuel burnup costs.

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* See LMFR, page 95.

2. Breeding

Very early in the development of the LMFR concept, BNL recognized that the concept had potential for breeding. The largest effort at BNL has been towards the development of a breeder reactor.

There are three possible designs which show promise of breeding: the single-region slurry fuel reactor presented as the low cost reactor in paragraph 1, above; the two-region design which utilizes a solution fuel in the core and a slurry in the blanket region of the reactor; and the internally cooled design in which the fuel is slowly circulated out of the reactor to remove fission products and a separate coolant is circulated through the reactor to remove the heat for the generation of useful power.

The chief advantage of the two-region reactor is that breeding may be obtained with a smaller reactor which permits higher specific power. If, in the long run, single-region reactors in the range of 1500 MWE become desirable, then breeding gains comparable to the two-region reactors can be obtained.

The chief disadvantage of the two-region reactor is increased complexity which adds to the number of technical problems that must be solved. From the breeding point of view, the problems are:

a. Adequate blanketing of the core. In a breeder, the blanket must adequately surround not only the core, but also the inlet and exit coolant nozzles. Neutron streaming losses from the nozzles must be reduced to a minimum. Radial blanketing of the nozzles involves an awkward mechanical problem.

b. The reactor graphite internals must be designed so as to keep intermixing of the two fluids within satisfactory limits established.

From the breeding point of view, the chief advantage of LMFR is the low cross-section of bismuth. A breeding gain of .05 can be attained. At present, the extent of fission product and xenon holdup within the graphite structure is still inadequately known and may affect the achievable breeding gain.

While additional complexity of design is certainly inherent in the internally cooled LMFR design, only in that design does the goal of a 10-year doubling time appear possible.

B. The LMFR Approach

1. The Concept

a. Fuel and Moderator of Basic Concept

The basic LMFR concept is the use of a liquid metal as the carrier (solvent or suspending liquid) for fuel and fertile material. Bismuth is used because of its low neutron cross section, low vapor pressure, comparatively low melting point, and satisfactory uranium solubility. Magnesium and/or zirconium may be added to the bismuth to aid wetting of slurries and to minimize corrosion mass transfer problems. The moderator is graphite and can be used uncanned in direct contact with the fuel. In the primary system the fuel is heated from 750 to 1050°F within the reactor; it is pumped from the reactor through a heat exchanger where it transfers the heat to sodium and back to the reactor.

b. Alternative Designs and Other Possibilities

There are many arrangements by which liquid metal fuels can be used to produce power. The two designs under consideration here are:

(1) The one-region slurry fueled reactor designed for minimum power costs, and burning U-235 and thorium with a conversion ratio of about 0.7.

(2) The two-region solution core, slurry blanket power breeder operating on the Th-U^{233} cycle.

Other interesting possibilities include: a burner for recycled plutonium with some conversion of natural (or depleted) uranium, and an internally cooled breeder with lowered inventory and a shorter doubling time.

c. Merits and Limitations of the Concept

The special merits of LMFR, in addition to those common to the three fluid reactors under consideration, are:

(1) High temperature and high thermal efficiency but without high pressure;

(2) Absence of stored energy (chemical or mechanical) which might scatter fission products;

(3) No gas production other than fission gases;

(4) Minimum thermal flux in components external to the reactor;

(5) Minimum criticality problems after a spill;

(6) Flexibility with respect to neutron energy;

(7) The melting point of bismuth is 525°F, which is sufficiently low to present no insoluble engineering problem while retaining fission products when it is in the solid state at room temperature;

(8) The use of bismuth makes it possible to use low cost materials such as unclad graphite and low chrome steel, with its long use experience;

(9) Under conditions where corrosion and mass transfer of the container material are acceptable, long term operation without chemical processing might be possible.

LMFR systems have the following limitations, in addition to those common to the fluid fuel concepts under consideration:

(1) The heat capacity of the fluid is lower and the density is greater, resulting in larger heat exchangers and higher pumping power requirement;

(2) There has been no demonstrated solution of the corrosion-mass transfer problem for the otherwise best choice of container material at $\Delta T = 300°F$.

(3) Since the fuel charge is solid at room temperature, equipment is required for preheating the system before charging.

(4) Because it is difficult to remove fissionable material, poison rods are probably necessary.

(5) The fuel itself has no moderating properties and moderator must be supplied.

(6) Since the solubility of thorium is low, breeding can be achieved only through the use of a slurry.

(7) Polonium is formed along with fission products.

2. The Development of the Low-Cost Power Potentialities

a. The Reference Design for the "Target" Plant

(1) Plant Description - Status of technology, problems to be solved, their relative difficulty and prospects of solution, for the following:

(a) The Fuel System and Related Container Materials Problems

Fuel Cycle

The core characteristics of this reactor design are established wholly by economic considerations resulting from an optimization study. The conversion ratio is well below the maximum attainable in this type of system and is an economic compromise of fuel burnup cost against vessel and inventory charges. The size of the core is a reconcilation of neutron leakage vs. vessel and inventory charges.

The U-235-thorium-U-233 cycle has been chosen for this design. The reactor is initially charged with 770 kg of U-235 and 23,000 kg of thorium as urania-thoria particles. This corresponds to 30 g. of thorium per kg of bismuth or approximately a 3 w/o slurry. The feed material is assumed to be U-235. The U-233 bred from the thorium is utilized in the reactor. As U-233 builds up in the fuel, U-235 required for criticality decreases to a minimum; then U-235 requirements for maintaining criticality increase as fission product poisons continue to build up until the fuel is removed from the system for decontamination.

Table X-1
Fuel Inventories in LMFR, kg

	Initial	Final	Average
Mass of U-235 in System	770	505	382
Mass of U-233 in System	0	656	542
Mass of Pa-233 in System	0	28	28
Total	770	1189	952

In the original reference design as proposed by the LMFR project the fuel is not removed from the reactor for chemical decontamination from fission products until the end of reactor life. Thorium is added to the system as needed to maintain a constant thorium concentration, and U-235 is added to compensate for fuel burnup and criti-

cality adjustments due to the buildup of U-233 and fission products. The variation of fuel inventory with time is shown in Figure X-A and is summarized in Table X-1.

The net burnup of fuel is a function of the conversion ratio which varies as shown in Figure X-B. A mass balance of fuel added and burned during life is given in Table X-2.

Table X-2
Fuel Feed and Burnup in 825 MW Single Fluid LMFR

Isotope	Mass at Startup Kg	Mass at 30 yrs. Kg	Mass Burned Kg	Mass of Fuel Added Kg	Total Utilization Burnup %
U-235	770	505	4,024	3,759	89
U-233	0	684	4,894		88
Thorium	23,000	23,000	5,700	5,700	20
Total	23,770	24,189	14,618		38

Chemical Processing

In this fuel cycle no chemical processing is performed on the fuel until 30 years of plant operation have been completed. The initial charge consists of fully enriched uranium and thorium contained in single solid phase particles as oxides dispersed in liquid bismuth. The solids concentration in the slurry suspension is approximately 3 w/o. Uranium and thorium are added at frequent intervals to maintain the proper concentrations for criticality, nuclear stability, and economy. After 30 years of operation without processing, the primary system contains 23,000 kg of thorium, 505 kg of U-235, 684 kg of U-233, approximately 950 kg of other uranium isotopes and approximately 5,500 kg of fission products plus corrosion products.

After plant shutdown, the fuel remains in the plant's dump tanks for radioactive cooling for approximately 200 days. The fuel is then transferred in small batches--still in liquid slurry--to the on site oxide slagging plant. Here the soluble fission products in the liquid phase are oxidized and, together with the uranium and thorium solids, are separated from all or most of the bismuth liquid phase, placed in shielded and cooled shipping casks for shipment to a central thorex-type aqueous processing plant for recovery of the fissionable isotopes. While the bismuth is not completely free from radioactivity, it should be sufficiently pure for recycling to another LMFR system.

FIG. X-A
FUEL INVENTORIES

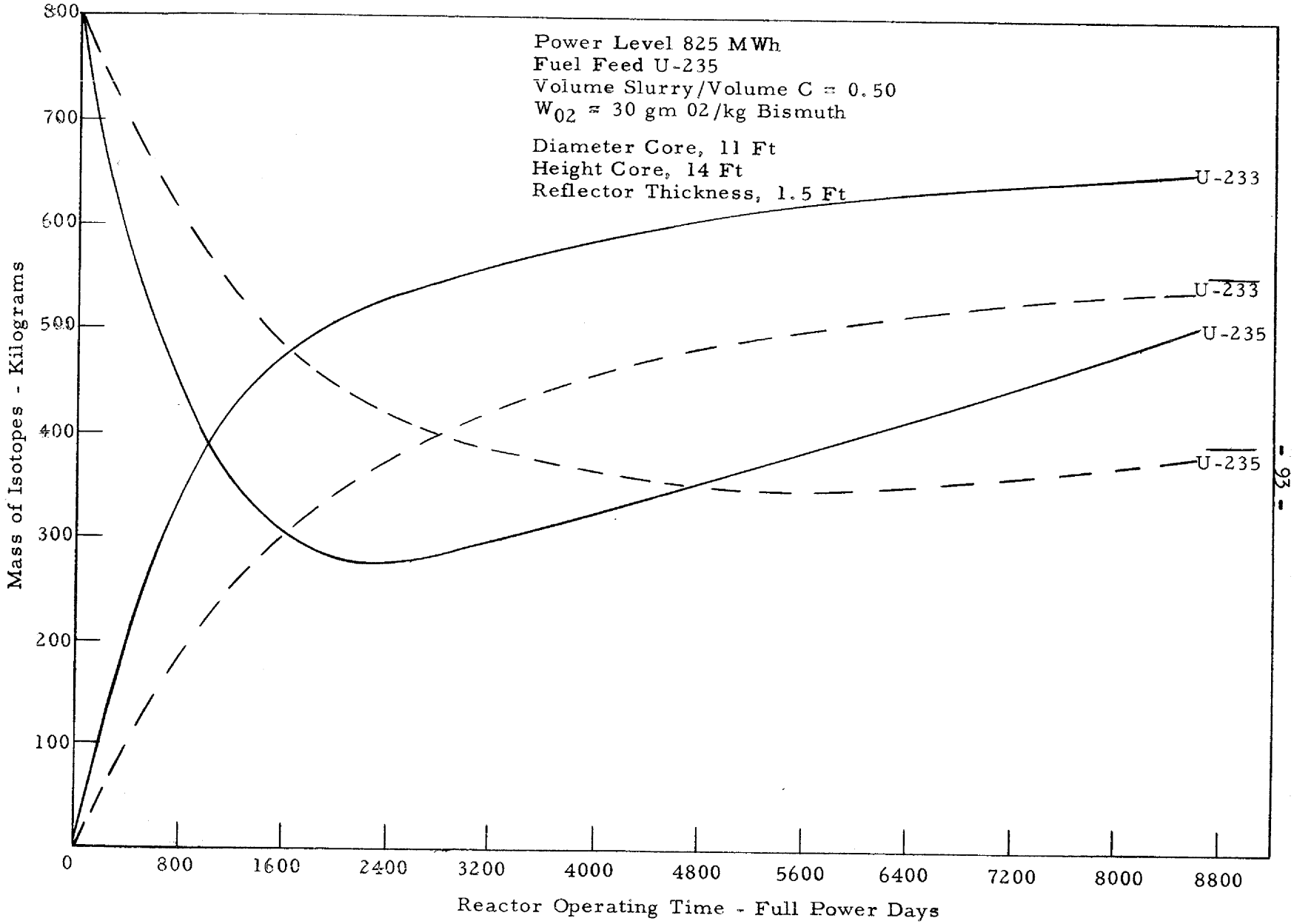
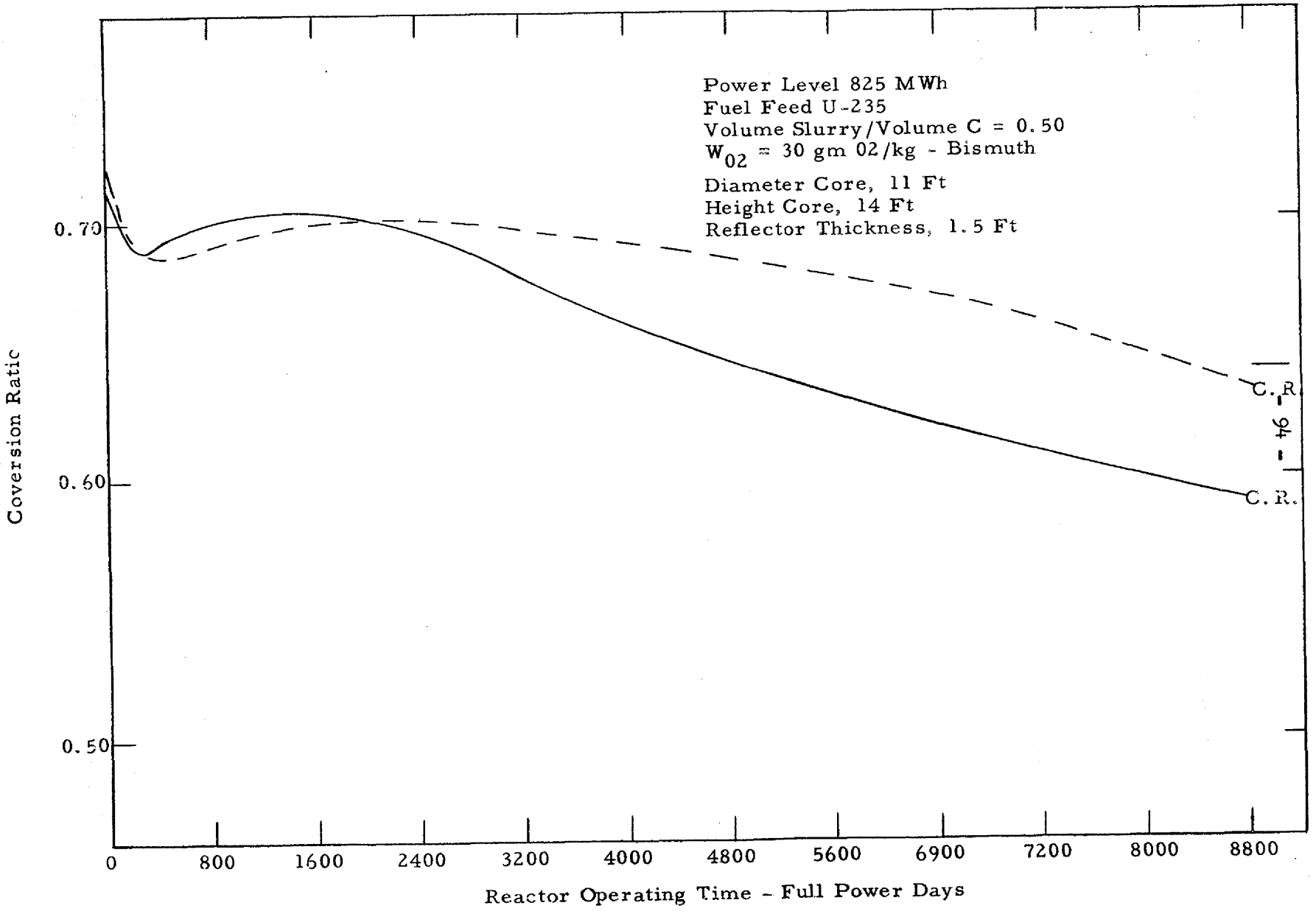


FIG. X-B

CONVERSION RATIO FOR SLURRY FUEL LMFR AS A FUNCTION OF TIME



In contrast to the above assumptions, it was decided on the Task Force that the fuel would have to be processed at a rate of 5% per year, and the chemical processing costs were estimated on this basis.

The chemical process has not been developed or designed. (This is one of the important items on the R&D program.) In the opinion of the LMFR Project Director a suitable process would consist of the following steps:

1. Treatment of the ThO_2-UO_2-f.p.-Bi slurry with NaOH and/or air to remove wetting agents and float oxides to the surface of the bismuth.

2. Mechanical separation of the bulk of the oxide from the bulk of the metal by flotation, skinning, etc. (separation need not be complete).

3. Oxide with some residual bismuth is either (a) frozen and put in on site Thorex process, or (b) cast into cans and sent off site frozen.

4. Thorex process (if on site one cycle gives sufficient decontamination).

5. Blending of $Th(NO_3)_4$ and $UO_2(NO_3)_2$.

6. Denitration to form mixed oxides.

7. Dispersion in Bi (from step 2) with wetting agents, and return to Reactor System.

The slurry must be processed at a rate of 340 lbs. or 0.5 cu. ft. per day to handle 5% of the charge per year. All operations with recycled U and Th have alpha and gamma activity and must be handled behind shielding and in an airtight system.

The chief differences of opinion between the project and the Task Force are on the probable difficulty of engineering such a process and its cost.

In the opinion of the Project Director, steps 1, 2, 3, and 7 (above) at the required scale could be done in one laboratory hot cell sized 10 ft. x 10 ft. with services. (A spare cell would be desirable.)

Preliminary cost estimates by experienced persons at Union Carbide Nuclear Company agree with the project estimates that this process can be done for much less than the Task Force estimates.

There are some alternates available: If the reprocessing costs turn out to be higher than the value of the recovered materials (as in the case of this Task Force cost estimate), then it would seem economical to store the fuel. This can be done safely and cheaply in frozen bismuth.

Bismuth and thorium would not be recovered unless the process cost less than the value recovered. Enriched U-235 and U-233 could not be thrown away but it should be allowable to store them until future process development or larger batch sizes made the recovery economical. Such process development does not appear (to the project) to be very difficult. At any rate there would be an economic incentive to develop it.

Present Technology

The important areas of research and development center around the development of a successful slurry and the compatibility of fuel and container materials.

Slurry Development

The immediate objective is the development of an insoluble uranium-thorium bismuth slurry. Current technology leads to the following summary:

1. Densities of solid and liquid are nearly identical before irradiation. During irradiation the density of the solid will probably decrease.

2. Viscosity of slurry is calculated to be essentially that of bismuth. No measurements have been made at operating conditions.

3. Density separation cannot be estimated until inpile loop studies have been made. Caking due to loss of electrical charge on particles cannot occur because the metal conductivity prevents such a charge. The reactor design reduces the objectionable features of separation as much as appears feasible.

4. The principal problem in this slurry is proper wetting.

While the slurry program is in its early stages, the following experience has been obtained:

1. Argonne National Laboratory has prepared and circulated ThO_2 slurries in sodium, demonstrating a slurry in liquid metal.

2. Knolls Atomic Power Laboratory has successfully circulated an 8.0 w/o UO_2 slurry in bismuth in a 3/4" loop.

3. Brookhaven National Laboratory and the Babcock & Wilcox Company have prepared and circulated oxide slurries in bismuth on a small scale, and have determined that (a) ThO_2-UO_2 powders can be dispersed in bismuth with proper wetting agents and agitation, (b) dispersions once obtained are stable for many hours without agitation, and (c) development has progressed to the point where very small pumped loops are being operated.

Compatibility of Fuel and Container Materials

The basic construction materials for containing these solutions are low chrome steels. There is a definite problem related to corrosion. Metal tends to be removed at hot regions, where its solubility in bismuth is relatively high, and to deposit at cooler regions, where the solubility is lower. This leads to a mass transfer, effectively corroding the hottest regions and tending to plug channels at coolest regions. The problem is magnified by increasing ΔT between hottest and coldest parts of the system, and by increasing temperature.

By careful control of steel composition, and by the addition of zirconium to the bismuth, it has been possible to control the corrosion and mass transfer of chrome-molybdenum steels at a ΔT of 180°F. Temperatures were 945° and 765°F. Steel containing 1¼% Cr and ¼% Mo (typical C = 0.12%, N = 0.010-.015%) has shown about 0.1 mil corrosive attack in 18 months exposure at flow rates up to 14 ft/sec. There has been some more rapid attack of welds, which would have to be prevented before performance could be described as satisfactory for this material at this ΔT.

The reference material is Croloy 2¼. In most respects the behavior of this alloy is very much like that of the Croloy 1¼ mentioned above. Many tests at relatively low ΔT have shown quite satisfactory performance. One thermal convection loop has been run successfully for about 7 months with a ΔT of 260°F. However, only one loop has been run for as long as 11 months at a ΔT of 180°F or above. This test, after about 10 months at 18-°F ΔT showed localized attack; after another month of operation the area of attack had increased. Maximum penetrations were about 15 to 20 mils. It is clear that the corrosion of Croloy 2¼ remains a major worry, and that considerable work remains to be done before the problem is solved.

Graphite gives every indication of being stable and unreactive with the liquid metal. Absorption of bismuth occurs in presently available graphite to the extent of about ½ gram per c.c. It is possible that pre-impregnation with pure Bi would minimize buildup of fission product poisons in the graphite. This would also prevent (less worrisome) fission recoils there. The effectiveness of this treatment cannot be predicted.

Erosion by slurries might not be a problem due to the density match of solid and liquid. It has not yet been evaluated experimentally.

(b) Reactor Plant

The reactor vessel contains a core and reflector assembly having a diameter equal to the height of 14 feet. The assembly is constructed of large graphite pieces machined to form a cylinder. The desired fuel content in the core is obtained by drilling vertical holes or channels having the proper size and spacing in the graphite matrix. The core is surrounded by a 1.5 ft. reflector containing only enough fuel channels to provide cooling of the graphite. The large graphite pieces are held tightly together with six circumferential temperature compensated metal bands.

The fuel-coolant stream (UO_2-ThO_2 slurry in bismuth) enters the bottom of the reactor at 750°F, flows upward through the graphite where criticality is achieved, and exits from the top of the core at 1050°F. The fuel enters into and exits from hemispherical plenums located at the top and bottom of core. Also a cylindrical space above the core allows for expansion of the fluid and a space for the expansion of the fluid and a space for the control rod drives.

The reactor vessel is constructed of Croloy 2¼ (low chrome alloy steel) with the main shell course having a thickness of three inches. The design pressure is 150 psi and the design temperature is 1100°F.

The reactor vessel is surrounded by a close-fitting containment vessel 16 ft. 6 in. ID and 1¼ in. thick. This vessel contains any spill of radioactivity out of the reactor vessel and serves as a container for gas used in preheating the vessel during startup and removal of decay heat after shutdown.

Gamma and neutron heating of the reactor vessel wall does not appear to be a serious problem in this reactor. The inside wall of the vessel is cooled by the fuel-coolant solution passing upward through an annulus approximately 1.5 in. thick external to the reflector graphite. Thermal shields appear unnecessary. This is due

to the excellent gamma shielding properties of the bismuth in the core and the neutron attenuation in the large graphite assembly.

Thermal stresses in the graphite have been roughly evaluated and do not appear to present a problem.

A possible problem is the regulation of the coolant flow upward along the fuel annulus between the graphite reflector and the reactor vessel. A hydraulic mockup is being built and tested.

(c) Primary System

System Design

The fuel stream enters the bottom portion of the reactor vessel at a minimum bulk temperature of 750°F, flows upward through the core where fissions within the fuel cause the fluid to undergo an average temperature rise of 300°F; the maximum average outlet temperature is 1050°F. Upon leaving the core, the fluid passes upward to a degassing area where volatile fission products are removed from the fuel stream. The reactor discharge is a header which splits the fuel flow into three primary heat transport loops.

Each primary system loop consists of two 20-inch pipes between the reactor and the pump, one 28-inch pipe between the pump and heat exchanger, and two 20-inch pipes between the heat exchanger and the reactor.

From the degassing area discharge header, each fuel stream flows to the suction of a variable speed, centrifugal pump designed to deliver approximately 17,500 gpm at 20 feet of pumped fluid. Each pump requires 1250 bhp. The pump is bayonet mounted through a steel shield plug which is in turn mounted in the canyon floor. This plug also serves as the mount for the intermediate heat exchanger. This arrangement permits semi-direct maintenance of the pump.

To prevent flooding the upper parts of the pump, the liquid level in the pump barrel is the same as the liquid level in the degassing area at zero flow. At full flow it is about four feet lower than the degassing area level. Pressures in the degassing area and in the space above the pump level are equalized at all times. A check valve, integrally attached to each pump discharge, is removable with the pump internals for maintenance. Purpose of these valves is to prevent thermal shocking in the reactor outlet with colder back circulating fluid from the reactor bottom in the event a pump stops.

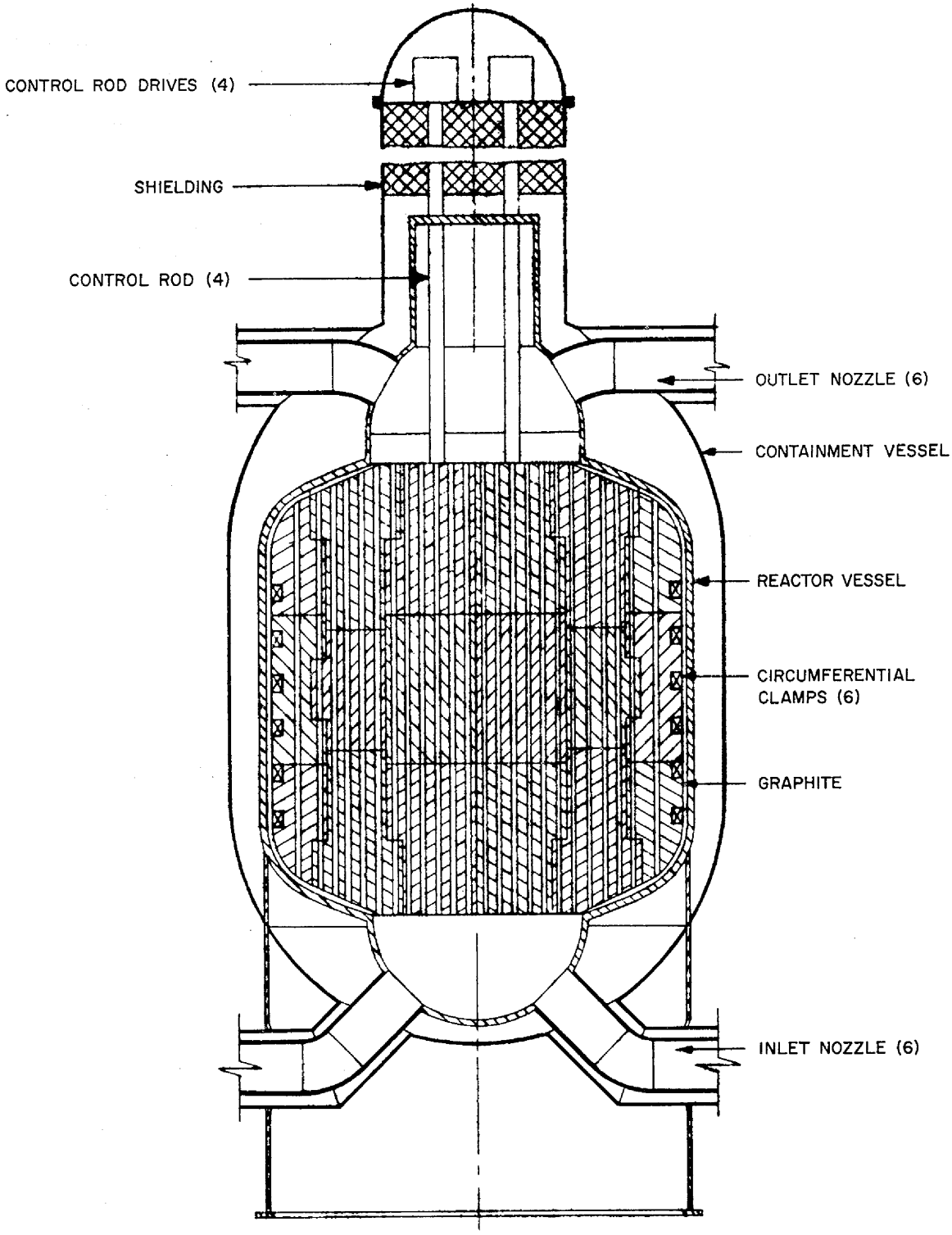
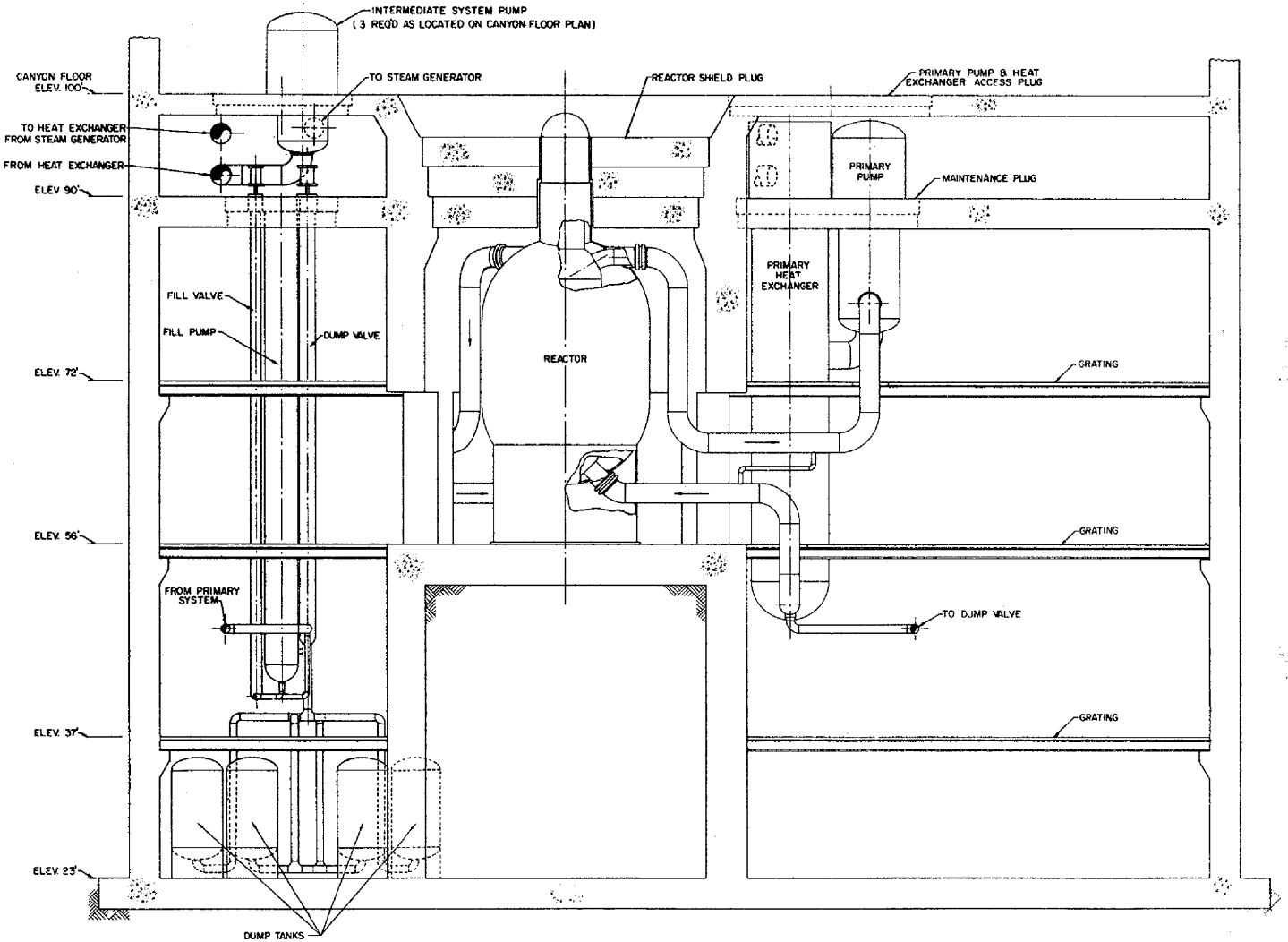


Fig. X-C—Single Fluid Reactor for 315 Mw(e) 3-loop Plant.



3 LOOP 315e MW
LMFR
SEMI-CONTACT MAINT.

Fig. X-D

From the pump discharge the fuel stream flows to the shellside of a bayonet mounted, semi-contact maintainable, intermediate heat exchanger. It is a counterflow, floating head unit containing 3000 3/4" OD tubes swaged to 5/8" OD at each end. The effective tube length is 25.5 feet. The primary fluid enters at the middle of the shell, passes downward around the tube bundle and exits at the bottom of the shell, and returns to the bottom of the reactor. The secondary fluid, sodium, enters at the top of the exchanger, passes down through a pipe centered on the vertical axis of the exchanger to a plenum beneath the tube bundle, then up through the tubes and exits through an annular space surrounding the inlet sodium pipe.

Each primary loop is provided with six dump tanks equal to the loop plus a portion of the reactor volumes. The tanks are safely sized and are provided with agitators to prevent settling of the slurry. One dump line with a bayonet mounted, semi-contact maintainable valve, connects the low point of each loop with the loop dump tanks. The primary loops are filled from the dump tanks by means of bayonet mounted electromagnetic pumps.

A number of design problems remain to be investigated, e.g. cavitation, rotating seals, bismuth lubricated bearings.

Many test loops have contained bismuth to bismuth heat exchangers. A five megawatt (design) heat exchanger has been manufactured and installed in the BNL four-inch loop. A bismuth to sodium heat exchanger is not considered to require a large development program. The development program being conducted in this area concerns seals between the inner and outer shells to protect personnel during replacement. Such seals are required for all semi-contact maintained equipment.

Heating and Cooling

Heating and cooling gas systems are provided for reactor, primary system, and dump tanks. Each system circulates helium through the containment annulus. Gas is heated in an electrical furnace and cooled through heat exchangers to water. The heaters, coolers, and blowers are all located in areas adjacent to the primary system. These components are arranged for remote removal in case of contamination by a primary system leak, but it is expected that most maintenance will be performed directly.

The total heating capacity of the electrical furnace is about 3 MW. The heat exchangers are capable of removing about 20 MW total. Different parts of the primary system will be serviced by different blowers, etc. but the subdivision has not been specified.

Containment System

All systems and components which are likely to contain fission products have close fitting double containment. The containment is all welded, and is helium leaktight. The containment is fitted with expansion joints. The space between piping or components and their containment is divided into several regions by bulkheads. A helium atmosphere is maintained inside the containment.

The basic function of the containment system is to contain leakage from the primary system. Close fitting containment was selected for this purpose for structural reasons, and because it could be used to heat and cool the system; to aid in detecting leaks; and to maintain an inert gas blanket around the primary system.

Off Gas System

The off gas system is based on continuous recirculation of gases (helium mainly) required to make up seal leakage, for operation of samplers and similar mechanisms, and for sweep of the reactor. The recirculation time is long enough that gas entering the primary system consists essentially of non-radioactive inert gases plus krypton 85.

The design of the system is based on the premise that all gaseous fission products with half lives longer than 9.8 seconds enter the gas phase. The kinetics of fission gas release from bismuth UO_2-ThO_2 slurry are not well known.

The distribution of heat in the off gas system is controllable by the flow rate of recirculated gas used in the plant. The total fission gas production rate is about 7.0 ft^3/day at operating temperatures. The estimated recirculating gas requirements for plant operation is about 6.0 ft^3/hour, so that this is controlling and is used to estimate the distribution of the volatile fission products in the off gas system.

The off gas system consists of a gas treatment loop, an operational gas sub-system, and a vacuum facility.

The gas treatment loop consists of a gas cooler, a gas particulate filter, a charcoal adsorption bed, a surge tank, a compressor, a holdup and decay tank farm, instrumentation, and connecting piping.

The operational gas sub-system is composed of a coarse gas filter, a surge tank, a vacuum pump, a compressor, and such control instrumentation, and piping as required. Gas required for reactor plant operation is drawn from the reprocessed gas storage tank, and

recycled into the gas treatment loop. The reactor sweep gas is drawn directly from the reprocessed gas storage tank, and is discharged from the reactor into the gas treatment loop.

The vacuum facility includes a gas cooler, a vacuum pump, storage tanks, and a stack for waste gas atmospheric disposal. This facility is required for the initial outgassing of the primary system, and the evacuation and transport of spent gases from the primary system during emergency operation.

Additional auxiliary systems are:

Inert Gas System
Shield Cooling System
Cell Cooling System
Raw Water System
Waste Disposal
Plant Ventilation
Storage Pool

(d) Intermediate System

The intermediate system, which also consists of three separate heat transfer loops, utilizes sodium as the heat transfer medium. All material of construction of the intermediate system, except the steam generator, is 2¼ chrome - 1 moly steel. The steam generator is constructed of type 304 stainless steel. The intermediate piping - 24 in. schedule 30 - is sized for a maximum sodium velocity of 17 ft/sec.

Sodium flowing at 22,100 gpm enters the tube side of the intermediate heat exchanger (which is the bayonet mounted, semi-contact maintenance unit previously described) at 680°F, flows downward through the central downcomer, enters the bottom tube header, and flows into the tubes. Sodium flows upward through the tubes countercurrent to the fuel stream and exits from the units at 1010°F. From the heat exchanger, the hot sodium flows to the suction of a variable speed centrifugal pump. Each intermediate pump is designed to deliver 22,100 gpm at 180 ft. head. Each intermediate pump is bayonet mounted through a shield floor in a manner that permits semi-contact maintenance. Bayonet mounting of the intermediate pumps is provided to minimize pump maintenance time since the radiation levels due to activated sodium in the vicinity of the intermediate pumps could prohibit contact maintenance of the pumps for a period of about three days.

From each pump discharge, sodium flows to the shell side of the steam generator. The steam generator is a U-tube, U-shell, "once through" type unit. The units consists of 1066½ in. OD tubes with an average length of 65 ft. The shell OD is 29 in. and the overall length is 68 ft.

Sodium flows counter-current to superheated steam, boiling water, and feedwater in the steam generator and gives up heat which produces 1,100,000 lb/hr of superheated steam at 2,250 psig and 1000°F.

From the steam generator sodium flows to the intermediate heat exchanger inlet to complete the cycle.

In addition to the components listed above, auxiliary components are necessary to obtain proper function of the intermediate system.

An expansion tank is located at the highest point of each intermediate loop. This tank serves as a cushion for pressure surges, a surge vessel for thermal expansion of sodium, and suction head for the pumps.

The lowest point of each intermediate loop is connected by pipe and dump valves to a sodium dump tank which receives the inventory of the respective loop. Dip tubes, which extend downward through the inlet downcomer of each intermediate heat exchanger, are provided for drainage of the heat exchangers.

A plugging indicator and a cold trap are provided to determine sodium oxide concentration and to maintain the oxide concentration at low levels.

The intermediate system will be heated by induction heaters.

(e) Control and Instrumentation

A load change will appear in the steam system as a change in throttle valve position and, therefore, a change in steam flow and pressure. The feedwater controllers at the inlet to the steam generators will sense these changes and operate to maintain steam pressure constant. The steam flow might also provide an anticipatory signal to the primary and intermediate system pumps to change their speed to suit the load.

The prompt and overall temperature coefficients of the reactor are negative. The present control concept utilizes the self-regulating properties of the reactor to compensate for small transients and a regulating rod to compensate for large transients.

Constant core inlet and outlet temperatures are maintained between 20% and 100% of full power by varying pump speed and adjusting regulating rod to compensate for delayed neutron changes and xenon poison effects. Below 20% of full power, a constant average temperature program regulates the reactor.

The reactor nuclear instrumentation system provides the reactor period and power signals necessary to actuate, overpower, or short positive period safety circuits, control signals necessary for reactor power control and the operation data necessary for the operation of the plant. The instrumentation system will consist of conventional startup, intermediate and power channels with the detectors placed peripherally outside the secondary containment in a relatively low ambient temperature (200°F). The instrumentation will provide nuclear data for safe reactor startup when an artificial neutron source is used. Dual channels will be used to increase the reliability of the equipment and to provide a double check on the neutron power or period indications.

Experience to date indicates that bismuth has not introduced any unique problems in the use of non-nuclear instruments for liquid metals. The status of technology is:

(1) Tests show that Croloy-2¼ Cr-1 Mo sheathed thermocouples should be satisfactory for LMFR service.

(2) Diaphragms in contact with bismuth are used in both pressure sensing and level devices. Diaphragms are currently undergoing tests for reliability and maintenance.

(3) Level indicators are now on test including differential pressure, J-probe, and float types.

All of the fuel addition to the primary system of the LMFR will be made by way of a single fuel addition mechanism. This mechanism will consist of a one-hundred gallon agitated tank into which the primary system fluid can flow by gravity and be forced out by helium pressure. The UO_2-ThO_2 powder will be introduced into this tank by way of a lock which will serve several purposes.

Fuel sampling in the LMFR will consist of a mechanism which obtains a sample of the slurry fuel from a by-pass loop containing an EM pump and a sampling pot.

(f) Fuel Handling and Processing Systems

The most economical chemical processing cycle time for single-fluid LMFR design is very long, e.g. twenty to thirty years of continuous operation before any processing. At the end of this period, the entire reactor charge is processed for recovery of uranium and bismuth on a batch basis. The chemical separation process consists of two major phases:

(1) On site plant - removal of slurry from the reactor, separation of fuel and thorium from bismuth, storage for radioactive cooling and preparation for shipping.

(2) Off site plant - chemical separation of the uranium and thorium in an off site, central processing plant using aqueous chemistry (thorex process).

For the on site plant, batches of slurry are taken from the reactor and are oxidized by passing a stream of oxygen through the melt or by slagging with caustic. After oxidation, a liquid-solids separation step is required.

(g) Special Maintenance Facilities & Problems

Major maintenance facilities in the LMFR plant are the following: Overhead crane, storage pool for radioactive components, decontamination cell for mobile equipment, hot cell facility for repair and replacement of pump parts and repair of other small radioactive equipment, maintenance tools, i.e. seal weld cutters, welders, remote viewing and portable lighting equipment and mobile trucks.

The design of the overhead crane is such that equipment subject to radiation damage and with a high repair frequency is placed in a shielded room, the crane is remotely operated from a central control room.

The major maintenance tools will include seal weld cutters for contact welding on the primary heat exchangers and pumps and remotely operated pipe cutters and welders. A prototype of a remotely operated welding machine has been built and has undergone tests with encouraging results. Remote viewing will be accomplished by the use of portable lighting equipment, TV units, and periscopes. The mobile equipment will consist of remote controlled mobile manipulators mounted on remote controlled life trucks, and remote controlled tractors. The mobile manipulator is undergoing performance studies. The remote controlled tractor is commercially available.

(h) Spare Equipment Requirements

Spare equipment for the reference plant was selected based upon expected failure frequency and time required for procurement.

Spares for Class I failures (72-hour shutdown) consist of primary and secondary pump rotating assemblies, intermediate heat exchanger tube bundles, dump valve operators and internals, instrumentation, and other items which are designed so that they can be rapidly replaced.

Spares for Class II failures (1-3 week shutdown) are limited to a steam generator, sodium piping, and insulation.

Spares for Class III failures (extremely long down time) include spare casings and containment for primary and intermediate pumps, IHX casing, dump valve, sampler and fuel addition casing, and spare primary piping and insulation.

A miscellaneous spare equipment inventory is included for such items as pump impellers, bearings, check valve plates, insulation, sodium filters and electrical parts.

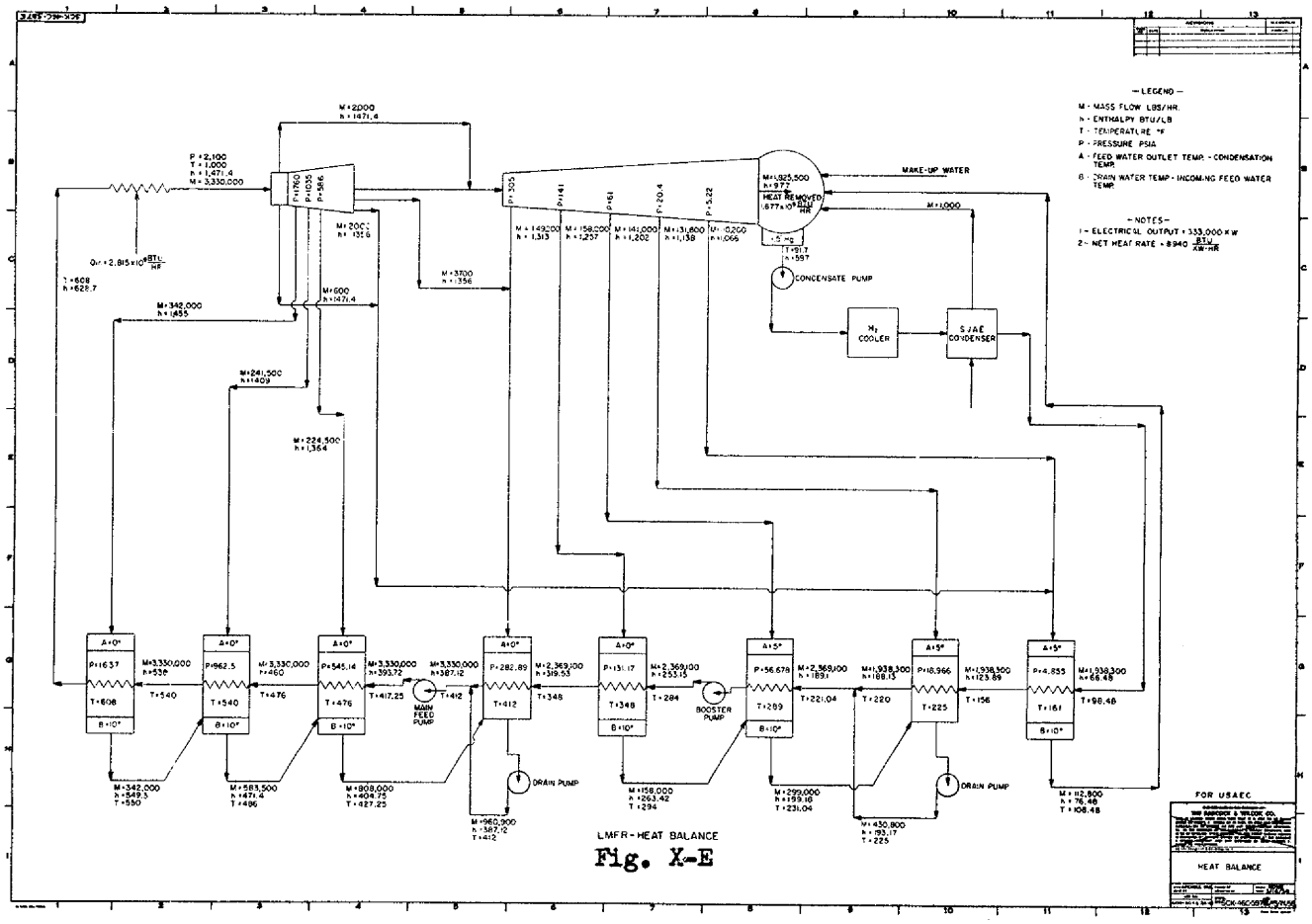
(i) Turbine System

A non-reheat, 2000 psi, 1000°F unit was selected for this application. The non-reheat feature was based on economic studies performed for BAW-2, which indicated lower costs for non-reheat, based on constant net electrical output. No attempt has been made to re-optimize this system based on today's prices; such a study may indicate that a reheat cycle may produce slightly cheaper electric power. The temperature for the system was based on once-through steam generator stability considerations. A heat balance is shown in Figure X-E.

About 18,000 KW of electrical power is used for the various pumps and auxiliary systems in the plant, making the net output 315,000 KW. Therefore, the net heat rate is 8940 BTU/Kwh which corresponds to an efficiency of 38.2 percent.

(j) Electrical System

The primary power supply is expected to be 34 KV/incoming to a 20,000 KVA three winding transformer, 34 KV - 4.16/4.16 KV with protective switchgear. The 4160 volt main distribution center will supply electric power directly to the large motors of 200 HP capacity and above, through appropriate switchgear. A 480/277 volt



LMFR - HEAT BALANCE

Fig. X-E

power center will be a double end feed unit from the 4160 main distribution center to furnish power for motors less than 200 HP.

To provide a reliable source of electrical power during emergency conditions, a diesel generator unit will be used as the auxiliary power supply. A constant power supply will be used for all critical instrumentation and control loads. Emergency lighting, alarm bus, PA system and battery vents will be powered from the constant power supply where safety of personnel is essential. A feeder from the 4160 V - main distribution center will be stepped down to 277/480 V - Wye, to provide 277 volt fluorescent lighting in all general areas. The high bay or reactor canyon will have mercury vapor fixtures. The 480/277 V plant distribution is used to (1) reduce fault currents, and (2) reduce conductor sizes. As other voltages are required they will be supplied locally by dry type transformers.

Induction heating will be used on the primary and intermediate systems for heatup. Present estimates indicate that 1625 KW will be required.

The turbine generator output will originate from two parallel generators each rated at 196,500 KVA with the voltage rating of 16,000 volts. The excitation voltage is expected to be 275 volts with a standby exciter generator. These generators will be the hydrogen cooled type with the coolers within the generator housing and designed to operate at 95°F. The generator output will be transmitted after a stepup transformer to provide a voltage of 115 KV through the usual protective switchgear. A single transmission line will be used to and from the plant site. From the main generator bus, power will be used through a station service transformer to energize a station auxiliary bus, making the generator output available within the plant.

(k) Buildings and Structures

The nuclear portion of the plant will be housed in a high-density concrete structure with the radioactive systems below grade where plant site terrain permits. The above grade "canyon" will be serviced by a remotely operated overhead crane. Wherever possible concrete will serve the double function of structural support and biological shielding. The plant will be arranged for simplicity of system arrangement and economy of piping and space, utilizing a semi-contact maintenance philosophy for radioactive systems.

The non-nuclear portion of the plant will be of standard construction, housing the turbine generator and auxiliary systems,

and the supporting facilities for administration, operation and maintenance of the plant. Plant arrangement and costs were developed in conjunction with a consulting architectural-engineering firm.

(2) Reactor Operating Characteristics

(a) Startup, Shutdown, and Load Change Behavior

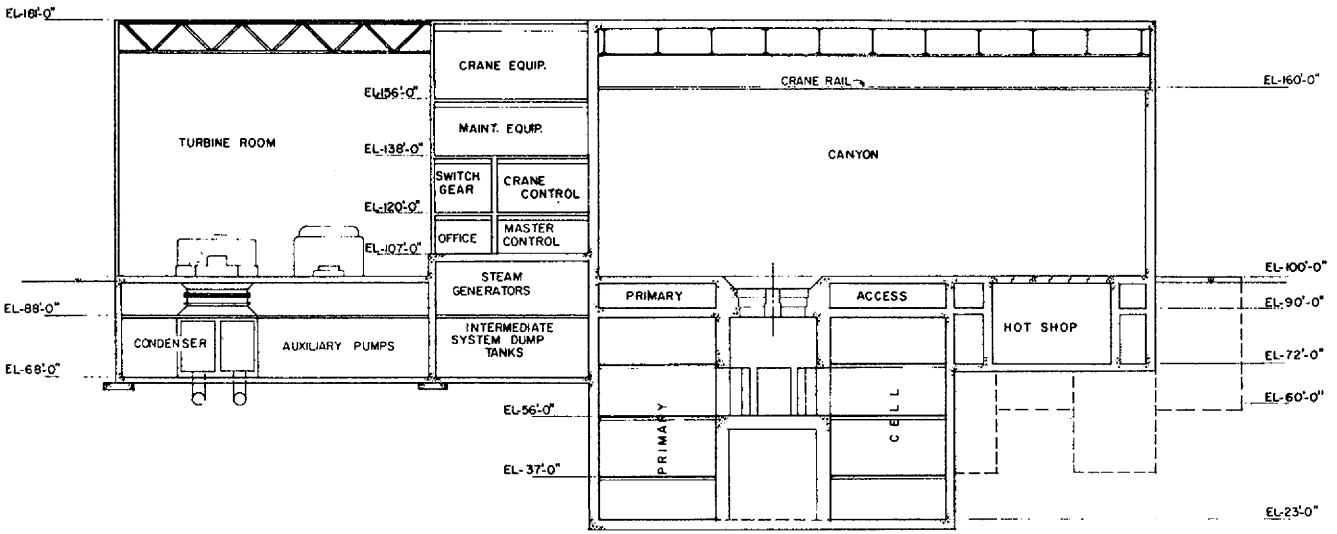
In the startup of the plant it will be necessary to preheat the primary system to 1000°F. This preheating of the reactor vessel, primary pumps and heat exchangers, all primary piping, and dump tanks will be carried out prior to filling the system with molten fuel fluid.

A limited preheating system which would consist of electric heaters will be needed on the secondary sodium system. The operational procedure for the steam portion of the plant would be similar to a conventional fossil fuel fired plant. The time required to bring the entire plant from a cold condition to one of carrying load on the turbo-generator will be largely determined by the time required to preheat the primary system. During normal shutdown the reactor decay heat is removed by generating steam and dumping to the condenser. Once decay heat is less than the system radiation losses the temperature coefficient will maintain the reactor just critical, with the slowly circulating system at the average operating temperature (900°F).

The reactor is equipped with a control rod system and variable speed pumps both on the primary and secondary loops. This system is proposed because of the limited knowledge of the transient response of the system to various disturbances.

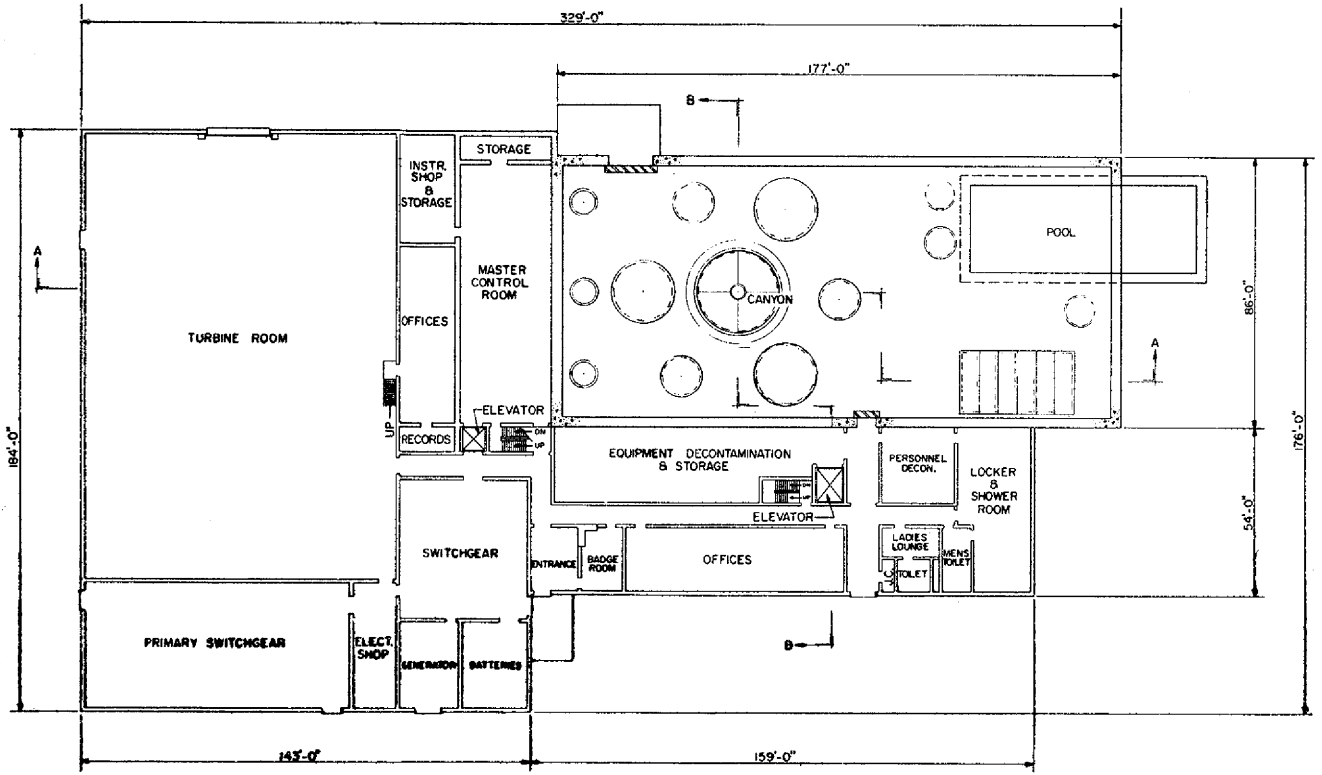
Power is varied from 0 to 20% of full load by operating the pumps at 20% of full load speed and gradually increasing the delta T across the core to 300°F. The time required to make this change is determined by the allowable rate of temperature increase in the container materials. Once 20% of power is attained, the reactor may be rapidly raised to full power by increasing pump speed and adjusting the regulating rod to compensate for loss of delayed neutrons.

Fuel sampling on the LMFR would consist of a mechanism which would obtain a sample of the slurry fuel from a by-pass loop. The samples would be transferred to a hot cell for analysis of uranium, thorium, and additives. The frequency of sampling will be determined by operating experience. In the control of the inventory of the liquid metal slurry, particularly when filling or dumping the primary system, it may be necessary to have the dump tanks act as weigh tanks. The use of a combination weigh and dump tank would



LONGITUDINAL ELEV. SECTION "A"-"A"
3 LOOP 315e MW
LMFR BLDG
SEMI-CONTACT MAINT.

Fig. X-F



GROUND FLOOR PLAN
3 LOOP 315e MW
LMFR BLDG.
SEMI-CONTACT MAINT.

Fig. X-G

give an indication to the operator of the amount of liquid metal in each tank and therefore his system inventory.

(b) Dynamic Characteristics

No special problems are foreseen.

(c) Instrumentation Problems

No significant nuclear or reactor instrumentation problems are anticipated.

(d) Hazards

An investigation of possible methods of injecting reactivity into this reactor has been reported. All nuclear accidents which can be postulated based upon realistic conditions can be handled by the control system without damage to the power plant or release of radioactive material to the environment. The maximum credible accident in this system is considered to be a rupture of the primary system spilling radioactive fluid into the close-fitting secondary containment. The only fission products which can escape are the volatiles and trace amounts of volatile halides and metallic fission products which leak through the containment into the cell and up the stack.

Dose calculations under the most unfavorable weather conditions predict lung doses due to iodine and lifetime dose due to Po-210 well below that normally considered as a maximum permissible emergency dose.

The absence of stored energy (due to low vapor pressure) the absence of any known potentially significant exothermic chemical reactions (such as metal-water reactions), and the inherent inability of this system to inject sizeable reactivity additions into the core should make this reactor extremely safe and immune to major accidents which would provide a serious radioactive hazard to the environment. This should allow and encourage the construction of this reactor near large centers of population.

(e) Containment System

All of the equipment in the primary system and some of that in the primary auxiliary system is enclosed in a close-fitting outer pipe or other metal barrier. Helium gas is passed through this annulus to heat or cool the components during abnormal operating conditions. The containment annulus is segmented and positive seals

are provided between segments in an effort to minimize the spread of radioactivity in the event of a leak into the annulus.

The design conditions for the secondary barrier are 35 psig and 1100°F.

(f) Other Fuel Handling and Design Problems

Aside from the fuel addition system and the fuel dump system described in this report, no other fuel handling system is needed throughout plant life. There are no requirements for deconcentration of fuel, the replacement of the fuel carrier, or the chemical processing of fuel for the decontamination from fission products. At the end of plant life (30 years), the fuel may be handled for fissionable isotope recovery as described elsewhere.

With regard to pumping a slurry, the addition of 1.5% to 3% by weight solids to liquid bismuth should not introduce any critical problems. Addition of slurry particles can conceivably introduce pump bearing plugging and scoring problems where fine clearances are required. Such problems may be alleviated by using hydrostatic bearings which do not require fine clearances, or eliminated entirely by using overhung shaft pump designs which do not require a submerged bearing. This question will be largely settled prior to LMFRE-II operation.

(3) Maintenance

In the reference design, the reactor, primary heat exchangers, primary pumps, and dump tank system are located in a compartmentalized canyon. A heavy concrete roof covers these compartments and has removable plugs over the different pieces of equipment. This roof forms the floor of the primary system access room and is at grade elevation. A remotely operated crane services this area.

The type of maintenance employed in this design can be called dry semi-contact, that is, using both contact and remote methods for the primary system. Chemical or other decontamination methods are incapable of removing sufficient contamination to allow direct maintenance on the primary system.

Major components of the plant are considered to last the life of the plant - these would include the reactor vessel, core graphite, primary piping, and dump tanks. Other components in the plant would have less than plant life and would have to be maintained. Equipment like the primary pumps and heat exchangers would be operated until they failed, at which time they would be removed and replaced with

a new component. Some contact maintenance will be possible on equipment mounted above the biological shielding, this would include the electric motors for the primary pumps, control rod drives, and valve operators for the dump valves.

In the operation of replacing a failed primary pump, service connections such as power leads, instrument leads, gas lines, and water cooling lines would be removed manually. A seal weld cutter, which is positioned manually, would then cut the seal welds. A component transfer container, which is equipped with a system of gate valve locks, would then be used to remove the pump. The actual removal of the pump from its sump would be done remotely due to the substantial gamma activity.

The replacement of a primary heat exchanger due to a tube leak will be accomplished by contact means either after waiting for the induced Na-24 activity to decay or after draining and flushing. The exchanger seal weld is cut by positioning a cutter manually, this is then followed by the use of transfer container to remove the tube bundle. The arguments for this method of maintenance hinge on the fact that the fuel carrier is a solid at ambient temperature, therefore, there is no danger of spills and it is felt that aside from gamma activity most fission products would be retained in the solid bismuth. However, there would be activity from fission products on the free surfaces of the bismuth. It is expected that repairs will be made on failed primary pumps in the hot shop. The LMFR study has chosen to dispose of failed heat exchanger tube bundles rather than attempt to repair them in a hot cell. This choice might be changed in the future as more detailed information is obtained on the investment required for tube plugging equipment and hot cell facilities needed for repair.

The need of a gastight outer container around a piece of equipment complicates the problems of maintenance. The greatest difficulty will arise when piping in the primary system has to be cut to remove a major component. Although this procedure has been studied in some detail and methods have been proposed to carry out this operation, there are a large number of problems that have yet to be resolved before practicality is established.

All maintenance on the dump tanks would be carried out remotely. The possible weak points in the dump tank system are failure of the dump valves, electromagnetic pumps for filling primary system, and overheating of the dump tanks due to decay heat from within the molten fuel. Alternate methods are available if proposed design is proven impractical.

The off-gas system, because of the high activity, would have to be maintained remotely. Because of the possibility of fission product gases leaking into the system in the reference design, the equipment for the helium preheating and cooling systems, e.g. blowers, electric heaters, and cooler would be located in a limited access area of the plant and sealed in individual containment vessels. Maintenance on this system's components would be by direct contact means when the system is not in operation. If the components become too radioactive and cannot be deontaminated, they will be destroyed by remote means and replaced.

The fan drive motor for the cell ventilating system which would be mounted above the plug, would be maintained directly. Work on the air fan and cooling coils would be done in the hot shop after the unit is removed from the cell. The air ventilation system for the hot areas of the plant would have to be located in a cell with means for decontamination of remote maintenance of the fans, motors, and filters. In the reference design, this system is outside the shielded part of the plant.

The instrumentation and the methods of maintenance have not been studied in detail. Some work has been done on a leak detection system for the primary system. This detection system is very important since all components and piping are enclosed by an outer sealed container. The detection system must serve two functions; show that a leak has occurred and give the location of the leak.

All maintenance work in the reactor, primary pump, and heat exchanger compartments below the main biological shield would be performed completely remotely. This work would include replacing the heat exchanger shells, pump casings, primary piping, dump tanks and their piping, and all instrumentation not accessible from the access canyon. The maintenance would be performed by mobile truck-mounted manipulators and television viewing equipment. The mobile manipulator truck, its manipulators and television equipment will be battery powered and radio controlled.

Two remote control trucks are under test to demonstrate the feasibility of this concept of maintenance.

The basic maintenance concept, using mobile trucks, will require extensive testing. In summary, the major problems of maintenance have been recognized and engineering work is underway to solve them.

b. Economic Appraisal

The Investment Cost Estimate for LMFR is given in Table VII-2.

The Power Cost Estimate for LMFR is given in Table VII-1.

Reductions in investment and power costs might be achieved in the future by refinements in design or as the result of technical developments. In particular, the following fields seem (to the project) to offer opportunities for eventual cost reduction:

(1) Increase in electrical generating capacity. Sharing on site services including chemical reprocessing and spreading cost over a larger power capacity.

(2) Elimination of intermediate system.

(3) Reduction in graphite cost.

(4) Use of U-233 or plutonium as fuel. (If recycled plutonium is available at a reasonable price, it can be burned in an LMFR and thus avoid difficulties of refabrication of radioactive solid elements.) A solution of plutonium and uranium in bismuth gives reasonable conversion without using a slurry.

(5) Simplification of plant and component designs.

(6) Elimination of control rods.

(7) Simplification of chemical process.

This is probably the most important item since the chief motivation of any fluid reactor is the opportunity to cut fuel cycle costs far below those of solid fuel reactors.

c. Development Program (Project Director's Appraisal)

The proposed LMFR development program includes two experimental reactors; an LMFRE-I (Ca. 5 MW) which is presently being designed to obtain basic information, and an LMFRE-II (Ca. 50 MW) which will be directed toward obtaining prototypic or advanced information requisite for the first large commercial power plant.

Prior to the fabrication of LMFRE-I, pertinent data relating to materials of construction and external reactor equipment must be obtained. To obtain this data, a comprehensive research and development program was formulated and begun by the Babcock & Wilcox Company

in January, 1957. This program covers physics, materials testing, instrumentation, component testing, and chemistry research. At the same time the Brookhaven National Laboratory is continuing its basic research on the LMFR and also in performing research specifically required for the first experiment where this work can be performed better at BNL than B&W.

Table X-3 lists the principal events for the overall LMFR program and the proposed schedule.

Table X-3
Program Schedule

Begin LMFRE-I construction	January 1960
Begin LMFRE-I precritical testing	May 1962
LMFRE-I startup & low power operation	September 1962
Begin LMFRE-I full power operation	February 1963
Begin advanced R&D for LMFRE-II	April 1963
Start design of LMFRE-II	September 1963
Begin fabrication of LMFRE-II	September 1964
Start precritical testing LMFRE-II	1968
Start design LMFR-I	1969
Begin LMFR-I pre-commercial operation	1974
Begin LMFR-I commercial operation	1975

Table X-4 summarizes the estimated costs for the complete development program. A more detailed analysis of the program may be found in Table X-5.

Table X-4
Cost Summary
LMFR Research and Development Program

A.	Basic LMFR-I Program	
	R&D required in order to build and operate LMFRE-I	$ 6,300,000
	R&D performed on LMFRE-I *	13,462,000
B.	Prototype LMFR-I Program	
	R&D required in order to build and operate LMFRE-II	11,700,000
	R&D performed on LMFRE-II *	45,200,000
C.	Basic Research on Advanced LMFRS	20,000,000
	TOTAL R&D Program	$96,662,000

* Includes conceptual, preliminary, final designs as well as construction and operation of experiment.

Table X-5
LMFR Research and Development Estimates - FY's 1959-1965

	1960	1961	1962	1963	1964	1965	Future Years	Final Year	Total Cost
One Region LMFR									
Slurry Fuel	1475	1285	1220	920	510	300	750	1968	6460
Materials									
Container	1910	1980	1880	1730	1180	750	900	1969	10,330
Graphite	820	1180	1030	710	450	280	250	1968	4720
Reactor Physics	340	300	400	400	500	400	-	1965	2340
Reactor Design Studies	650	600	500	400	650	650	-	1965	3450
Engineering Development									
Components	580	660	220	550	930	370	-	1965	3310
Systems	120	80	80	200	200	80	-	1965	760
Fuel Processing									
Removal of volatile f.p.'s	150	150	100	100	100	-	-	1964	600
Sep'n. of ThO_2-UO_2 from Bi	-	50	75	75	50	50	-	-	300
Operation & Maintenance	190	200	150	145	70	-	-	1964	755
Remote Maintenance	350	400	200	250	450	400	400	1967	2450
Total R&D for One Region LMFR:	6,585	6,885	5,855	5,480	5,090	3,280	2,300		35,475
Additional for Two Region LMFR:	-	-	225	750	1000	1300	2100	1968	5375
	6,585	6,885	6,080	6,230	6,090	4,580	4,400		40,850

The development program will include basic research in various fields:

A. Fuel

1. Slurry

a. Make-up Procedures - Object of this work is to determine best procedure for preparing UO_2-ThO_2 dispersions in Bi, best type and condition of UO_2-ThO_2, and best composition of Bi phase.

b. Chemical Stability - It will be necessary to determine the extent to which Th in UO_2-ThO_2-Bi dispersions undergoes chemical reduction, the objective being to find a dispersion that is for all practical purposes chemically stable.

c. Physical Stability - Work will have to be carried out with the objective of producing a UO_2-ThO_2-Bi slurry which will have a satisfactory low rate of phase separation, which will maintain its average particle size, and which will not precipitate UO_2-ThO_2 in flowing systems.

d. Radiation Stability - It will be necessary to develop a slurry which is satisfactorily stable in a fissioning environment. The influence of fission products will have to be determined.

e. Physical Properties - Viscosity measurements will have to be made; also, the tendency to wet and adhere to graphite and steel will have to be determined.

f. Sampling Procedures - It will be necessary to do further work on the development of slurry sampling procedures, particularly on radioactive systems.

g. Behavior in Flowing Systems - Both out-of-pile and in-pile loops will have to be run to study corrosion, erosion, and precipitation and to test components, instrumentation, and general handling procedures.

h. Instrumentation for Flowing Systems - A certain amount of research and development work will be required in this area.

i. Small Components for Flowing Systems - A certain amount of development work will be required in this area also.

2. Removal of Volatile Fission Products

The purpose of this work is to determine the behavior of the volatile fission products and their precursors in LMFR-type systems and, second, to establish the best method of their removal from the fuel.

3. Chemical Stability of Fuel-Graphite-Steel System

It will be necessary to determine the extent to which long-term operation will affect the fuel and the reactivity control. Also, the extents to which uranium and fission products penetrate the graphite must be known.

4. Release of Po and Fission Products from Molten and Frozen Fuel

This information is pertinent to safety and maintenance considerations.

5. Equipment Decontamination Studies

It is necessary to develop satisfactory techniques and procedures for decontaminating steel equipment which has held LMFR fuel with fission products or has been used for maintenance.

6. Recovery of U from Slurry Fuel

A satisfactory method will have to be developed for recovering the uranium in the fuel. A satisfactory method will have to be developed for separating the UO_2-ThO_2 particles from the bismuth. It is presumed that the Thorex process would be used for recovering the uranium in the oxide.

B. Materials

With the exception of possible effects of radiation, a 2¼ Croloy for the LMFRE-I suitable for 135°F ΔT operation for a minimum of two years has been obtained. The graphite for the core of the Experiment will have been completely tested in the next six months. The effect of irradiation on corrosion and the properties of the container material and graphite will be examined in three in-pile loops, two of which are undergoing out-of-pile checkout at this time. The results to date indicate that the construction of the E-I would not be held up by any material problems.

During construction of LMFRE-I, material tests will be continued to demonstrate the practicality of operating at a higher temperature rise across the reactor. It is planned to operate the first experiment at a temperature rise of approximately 225°F after initial operation at a ΔT of 135°F. This is expected to be carried out some time in 1963.

The material program will also conduct the prototype testing of the container material at the 300°F ΔT to be used in the second LMFR Experiment. The initial studies at the ΔT of 300°F will have been started before construction of the LMFRE-I. Where suitable, improved grades of graphite will be tested in the program outlined above.

As soon as the present screening tests at BNL and B&W justify it, 3/4" size out-pile loops followed by in-pile loops will be operated. Out-pile loops are available for immediate operation on a slurry. It is expected that by the time construction of the LMFRE-II is started in 1964, material suitable for 300°F ΔT and 2 years' operations will be available. Throughout this overall program additional studies will be carried out on the fundamental aspect of corrosion inhibition and irradiation effects as related to an LMFR. This work will probably be performed at BNL.

C. Reactor Physics

Exponential and critical experiments will be performed throughout the development period on the ranges of material composition and on the geometries of interest for the reactor experiment, prototype, and full-scale reactor.

Parallel with the above studies would be a programmatic investigation of the transient equations describing temperatures, reactivity, and component behavior of the reactor and heat extraction system. Since the behavior of the reactor experiment and prototype will no doubt suggest some changes in the calculational scheme, and since one can expect that the analysis will be in a constant state of improvement, the analytical investigation of transient behavior can be expected to continue at a reasonable steady pace up through and after the final construction of a breeder or full-scale power reactor.

D. Reactor Design Studies

Overall reactor and plant design studies will continue to be made as the LMFR development program proceeds.

E. Components

1. Pumps

Many liquid metals have been successfully pumped in the temperature and head range required for the LMFR. There is little difference between bismuth and lead and other heavy metals which have been pumped successfully. The development program is therefore aimed at modification of existing pumps for LMFR service and at extrapolation of pump sizes to those required for a large commercial power plant.

The most important factors to be considered are (1) the semi-contact maintenance requirements, (2) the physical location of the pump at the highest point in the circulating system, (3) the avoidance of cavitation, and (4) any modifications required in order to be able to pump a 1½ to 3 w/o slurry.

A large number of small pumps (15 gpm) have been used at BNL and B&W for pumping uranium-bismuth in out-of-pile loops. These pumps have operated very successfully in forced circulation loops; but, no attempt has been made to modify their design for use in an experimental reactor.

Two large-scale pumps (360 gpm) will begin demonstration runs in the very near future on the 4" Utility Test Loop at BNL. These pumps have not been designed for contact maintenance and therefore, it is expected that two 720 gpm pumps designed for contact maintenance will be operated prior to the LMFRE-I.

On present pump designs, shaft seals are located just below the pump motor in a gas atmosphere and their purpose is to limit fission gas leakage into the motor area. Various types of seals are being considered, primarily labyrinth, fluid, centrifugal and wet mechanical. Similar seals have been used successfully for sodium and salt pumps. Operation of the LMFRE-I should confirm or indicate modifications to seal design.

Cavitation is an important design area in LMFR systems because of bismuth density and plant arrangements which place pumps at high elevations to permit semi-contact maintenance. It is regarded, however, simply as a design parameter which must be determined for bismuth systems. Present pumps are conservatively designed from this point of view and can be designed even more conservatively, (greater suction head or lower rpm) if experimental data require such action. The characteristics determined for solution fuels are expected to apply to slurry fluids. Required data will be on hand in FY-1960 prior to operation of the LMFRE-I.

2. Intermediate Heat Exchangers

The use of bismuth does not appear to introduce any significant heat exchanger design problems. Experience has been obtained with liquid metal heat exchangers of many different designs. Some of them, especially those used for the Enrico Fermi plant and the EBR-II, are similar to the design configuration being proposed for the LMFR. The operating experience of these two plants will be available to confirm the LMFR design approach. Operation of the LMFRE-I exchanger will also provide useful information and indicate the direction of development on detail items.

The properties of slurries are not expected to have a major effect on heat exchanger design. Use of a slurry will require special attention to eliminate crevices and areas of low flow velocity. An engineering flow test slurry loop will be required at an early date to provide design information in this area for both pumps and heat exchangers.

3. Valves

There are three applications for valves in the LMFR primary system. Each pump has a check valve to limit back flow after pump shutdown. There are three dump valves, and three smaller fill valves basically similar to the dump valves.

The influence of slurry properties on dump and fill valve design should be small. They are designed for light service, i.e. each valve should be used only about 35 times during the life of its internals (5 years).

F. Maintenance

In addition to incorporating maintenance requirements in the design of the components, development of maintenance tools such as remote cutters, welders and mobile manipulators will be required. Sufficient work should be performed at an early date to prove, in general, the feasibility of the proposed maintenance method.

3. The Development of the Breeding Potentiality

a. The "Target" Breeder

Previous studies have resulted in an externally cooled two-region breeder reactor design. The core is approximately five feet in diameter and contains uranium-bismuth solution as the fuel. It is surrounded by a blanket region containing 10 w/o thorium suspended in liquid bismuth (as compared to 3 w/o Th in the one-region design). In both regions, the moderator material is graphite. (Figure X-C shows a cross-section of the reactor designed for this concept.)

Continuous chemical processing of the core fuel solution maintains a steady state fission product poisoning value consistent with neutron economy. For example, a core chemical processing cycle time of 76 days will provide a poison ratio of .03 for fission products other than xenon and samarium. The processing cycle for the blanket material is determined by the power generation and U-233 concentration desired in the blanket region. For example, if 10% of the total power is generated in the blanket system, the U-233 concentration in the bismuth is 230 ppm and the inventory is 48 kg. To obtain rapid reprocessing of the blanket, a "soluble" slurry of $ThBi_2$ in Bi offers the advantage of quick solution and reconstitution. The chemical processing of both core and blanket must be done rapidly and often and thus will probably require on site processing.

The breeding ratio expected in this design is approximately 1.026 and the total fuel inventory in both core and blanket systems is 417 kg of U-233.

This design provides sufficient neutron economy to meet the qualifications of a hold-your-own breeder. The characteristics of this reactor are shown in Table X-6 and the fuel costs are summarized in Table X-7. The breeding ratio can be increased to 1.05 at the expense of increased fuel inventory.

TABLE X-6
SPECIFICATIONS FOR EQUILIBRIUM OPERATION

Core:

Thermal Power	742 MW
Electric Power	283,000 KW
Diameter, Inches	61
Height, Inches	91.5
Fuel	U-233
v_{Bi}/v_c	1.22
N_{23}/N_{Bi}	620×10^{-6}
Mass of U-233 in system, kg	369
Total volume of fuel, ft^3	1915
Breeding ratio, overall	1.026
Chemical processing cycle, days	76
Volume flow rate through chemical plant, ft^3/day	25.3
Mass flow rate through chem. plant, g U-233/day	4,850
Average thermal flux in active core	2.2×10^{15}
Average thermal flux in core system	5.8×10^{13}

Blanket:

Thermal Power	83 MW
Electric Power	32 MW
Thickness, ft.	3.5
v_{slurry}/v_c	0.5
Slurry Content:	
Thorium 10% wt	
Bismuth 90% wt	
N_{23}/N_{Bi} (atom ratio) - 1190×10^{-6}	255
Mass of U-233 in system, kg	48
Mass of Thorium in system, kg	27,900
Total volume of fuel, ft^3	745
Chemical processing cycle, days	68
Volume flow rate through chem. plant, ft^3/day	11.5
Mass flow rate through chem. plant, kg of Th/day	410
Average thermal flux in blanket	1.6×10^{14}
Average thermal flux in blanket system	5.2×10^{13}

TABLE X-7
Fuel Cost for Two-Region Externally Cooled LMFR Breeder

Bi inventory	$ 440,000 yr.
Fuel inventory	353,000
Thorium inventory and burnup	145,000
Chemical processing	3,031,000
TOTAL	$3,969,000
Less credit for U-233	116,000
Net TOTAL	$3,853,000
	1.75 Mills/Kwh

The additional development problems (above those required for the one-region minimum-cost reactor) would be:

(1) Graphite Development - A two-fluid breeder requires a "wall" to prevent mixing of the core and blanket. Furthermore, this wall must not capture many neutrons. Graphite is a good candidate but more development of graphite properties and fabrication will be needed. The "wall" need not be entirely leaktight. Some mixing of core and blanket fluids can be tolerated but too much mixing is bad since it puts an extra load on the chemical processing. Graphite to metal seals will be needed and sliding seals have been tested and found satisfactory. Development is being done at present on the graphite.

(2) Design problems associated with extending the blanket around core ends will require prototype flow mockups.

(3) It is necessary to develop a method for rapidly processing the blanket fluid. A "soluble" slurry of $ThBi_2$ in bismuth appears to have advantages over the "insoluble" slurry of ThO_2 in bismuth with respect to chemical reprocessing. For this reason BNL has been developing a $ThBi_2$ slurry. A slurry of $ThBi_2$ dispersed in bismuth containing 0.5% Te and 0.025% Zr has been circulated at 1.5 ft. per second in a pumped loop operating in the temperature range of 400-500°C with a bulk delta T of 21°C and film drops of approximately 20°C for periods of up to 1500 hours without appreciable deposition of $ThBi_2$ or Fe in the cooled section.

No corrosion or erosion of the steel container or pump have been detected. Larger delta T, however, might cause particle growth or deposition in the cooled leg. Therefore the development work should be continued to increase the permissible temperature rise to that obtained across the core. Operation with a lower temperature rise across the blanket would lead to higher pumping costs and aggravated thermal stress problems.

(4) Chemical Reprocessing - Whereas only the volatile fission products would be removed from the fuel in the one-region reactors, all the fission products would be removed in the case of the two-region breeder. The breeder would have complete blanket processing.

Fuel reprocessing - processes for removing fission products from the solution (core) fuel have already been thoroughly studied at BNL.

In general, the additional R&D required for the two-region design is that necessary to develop the chemical processing for removing the FPS * and FPM ** fission products. Breeding dictates the removal of fission products to levels where their total reactor poisoning effect is no greater than about 3%.

The individual R&D items required are described briefly below:

Removal of FPS Fission Products

Chemistry - The chemistry of the fused-salt extraction process is pretty well established, although some mopping up remains to be done. Reaction mechanisms and rates need further study.

Instrumentation and Control - It is likely that the FPS's would be removed continuously; and for that, special instrumentation and control devices are needed.

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* FPS = fission products extracted in chloride salt, including alkalis, alkaline earths, rare earths.

** FPM = noble fission products not extracted in salt, including Ru, Rh, Pd, Mo.

Loops, Components, and Engineering - Loop N results will go a long way toward answering these kind of questions, but it is realized that further work will be needed, particularly on radioactive fuel.

Equipment Development - In order to carry out the above work a certain amount of development work on special equipment will be required.

Removal of FPM Fission Products

A process has been designed on the basis of small-scale laboratory work done at BNL and done for BNL by the American Smelting and Refining Company. This process is based upon a combination of salt extraction and zinc sludging. Further work will need to be done, particularly on radioactive handling procedures and engineering operations. The process is presumed to be based on batch operation.

Blanket Reprocessing - In the case of the $ThBi_2$-Bi slurry, processes for separation and recovery of the bred products, Pa and U, and for reconstitution of the slurry have been demonstrated on a laboratory scale.

In the alternate case of the ThO_2-Bi blanket, after separation of the ThO_2 solids from the Bi, the ThO_2 can be processed for U recovery and fission product separation by the well established Thorex process. But there is reason to believe that a cheaper non-aqueous process is possible. One which looks interesting is a modified fluoride volatility process. In this, the ThO_2 is dissolved in NO_2-HF and the U separated as UF_6 by the addition of F_2 or BrF_3. This assumes that the non-aqueous process would be sufficiently cheap, compared to the Thorex process, to warrant its complete development.

b. Internally Cooled "Short Doubling Time" Breeder LMFR

If the expansion of electric power is faster than can be supplied by "hold-own" breeder reactors, then a design which emphasizes the importance of doubling time which in turn places great importance upon high specific power and high breeding gain, will have to be provided.

Design experience on circulating fuel reactors indicates that relatively large coolant volumes external to the reactor core are required to provide remote or semi-remote maintenance procedures and a piping layout to withstand operational stresses. This inherent design feature suggests that the best method of reducing fuel inventory and improving doubling time is the use of an internally cooled reactor concept.

Preliminary Reactor Design

The core consists of an array of graphite elements made of square graphite stringers. Both fuel and coolant holes are drilled axially through the stringers and separately headered at top and bottom of core. Molybdenum fuel headers at both ends of the core were chosen in order to reduce the problems associated with thermal stresses in the graphite to metal connections.

The fuel consists of U-233 and thorium slurry particles suspended in bismuth with a thorium concentration of 50 grams per kilogram of bismuth. Almost all of the fission heat generated in the fuel stream is transferred to the pure bismuth coolant. Fuel is slowly circulated outside the reactor vessel in order to permit degassing and fuel addition.

The unit cell is determined by the characteristics of graphite. The maximum thickness of the graphite between fuel and coolant channels is determined by the allowable thermal stresses in graphite. The overall diameter of the core (10 ft.) is determined by the heat transfer surface required to transfer 825 MWH.

The core is surrounded on the sides by a 2.5 ft. thick blanket containing a 10 w/o thorium in bismuth slurry, pure molten bismuth as the coolant, and graphite as the moderator. The fertile slurry is continuously removed from the blanket for chemical processing and separation of U-233 for feed into the core. A graphite reflector, at least two feet thick, at top and bottom of the core is provided for neutron economy.

Preliminary calculations indicate that a doubling time of the order of ten years is feasible in this system from a physics standpoint.

The development of an internally cooled LMFR will involve many additional research and development programs. These are centered around the design of the reactor itself, since the external heat removal system can be the same as that presently contemplated for externally cooled LMFR's.

The problems in the reactor will be the development of graphite stringers of extremely high impermeability and an exceptionally leaktight graphite-metal joint which can withstand temperatures in the range of 1700-1800°F.

In addition, highly efficient, low cost, on site chemical processing must be developed.

With regard to the time scale on this part of the overall LMFR development program, the whole question of the importance of breeding is such that it should suffice to say that no development work in these specific areas should be started until there are more positive indications that a breeder of this type will be desired.

XI. AQUEOUS HOMOGENEOUS REACTOR PROGRAM

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XI. AQUEOUS HOMOGENEOUS REACTOR PROGRAM

A. Program Objectives

Development of aqueous homogeneous power breeder reactor systems which will produce economical nuclear power in central station installations is the objective of the homogeneous reactor program. Major emphasis is placed on development of power breeder reactors fueled on the thorium-U^{233} cycle. The program is founded on estimates that low fuel cost resulting from simplification of the fuel cycle by use of fluid fuels and from complete utilization of fissionable and fertile material is an important factor in the ultimate economics of nuclear power.

Incidental to the development of power breeder reactors, studies have been made of other aqueous homogeneous power reactors which utilize similar fuels and equipment and have all the advantages of the power breeders except that they have lower conversion ratios, and therefore poorer fuel utilization. Included in this category are U-235 burner reactors, plutonium production reactors, reactors fueled on the uranium-plutonium cycle and thorium-fueled reactors which have conversion ratios less than 1 and require a continuous feed of U-235 to make up the deficit. Depending on the particular reactor it may be relatively simple to build and operate; it may offer promise of producing high-priced weapons materials at low cost; or the fuel cost may be low in a period of plentiful and cheap fissionable material.

A continuing interest in alternates is warranted where they are logical steps in the development of large power breeder reactors and where, as the development progresses, they offer promise of producing power more economically than it can be produced in other types of power reactors. However, breeding with complete utilization of thorium in large power stations remains the primary goal of the work.

B. Advantages and Disadvantages

The major advantages of aqueous homogeneous reactors as compared with other fluid fuel reactors result from the favorable characteristics of heavy water as a moderator and fuel carrier. Use of heavy water as a moderator and uranyl sulfate, uranium oxide, and thorium oxide as fuel compounds makes possible design of circulating fuel reactors which have the lowest loss of neutrons by leakage and parasitic absorption and, therefore the highest conversion ratio and

breeding gain. These same designs have specific powers equal to or greater than those of other fluid fuel reactors with resulting shorter doubling time in breeder reactors. The fuel concentration in aqueous systems can be altered by simple physical methods. This characteristic facilitates both control of the reactor by control of fuel concentration, separation of fuel and fission products from carrier for processing and washing of the reactor to remove fuel before performing maintenance operations. The moderator circulates with the fuel, is continuously purified from fission products and although subject to radiation damage, the damage products, D_2 and O_2, can be recombined by catalysts in the fuel or in external systems. No special provisions are required to prevent the fuels from freezing so many of the equipment and fuel handling problems are simplified. There is no requirement for maintaining an inert atmosphere in contact with the fuel.

Disadvantages of these reactors are also related to the characteristics of water and the fuel compounds. Fuels which are now in use have major faults. The uranyl sulphate solution fuel is corrosive and its stability decreases with increasing temperature. The thoria fuel is erosive, and settles unless it is agitated continuously; no methods are presently envisioned which show promise of separating bred uranium from the thorium without dissolving the thoria particles. Corrosion of materials in the reactor core is increased by radiation. Special precautionary measures must be taken in the design and operation of the reactors to prevent the moderator decomposition products, D_2 and O_2, from introducing a serious explosion hazard. Aqueous systems operate at relatively low temperature so the thermodynamic efficiency of the power cycle is low. The vapor pressure of the water is high at the operating temperature of the reactor so the radioactivity is more difficult to contain. Care must be taken to prevent the expensive heavy water from being contaminated by light water.

C. Reactor Types

Three types of reactors are considered in this evaluation. Two are circulating-fuel, two-region, power-breeder reactors being developed at the Oak Ridge National Laboratory. The third is a one-region reactor similar to that proposed by the Pennsylvania Advanced Reactor Project. One of the two-region breeders has a core of dilute solution (5 to 10 g/liter) of uranyl sulphate in heavy water surrounded by a thoria slurry blanket of 750 to 1000 g Th/liter in heavy water. The other has a thorium oxide - uranium oxide slurry of 200 to 300 g Th and 15 to 25 U/liter in heavy water. U-233 is produced in the blankets of the two-region reactors and permitted to reach a steady state concentration of 1 to 5 g U/kg Th before processing, the concentration depending on the cost of processing, the value of U-233 and the importance of a minimum doubling time. The one-region reactor is essentially the two-region slurry-core reactor without blanket.

Fuel for the reactor would be a thorium oxide - uranium oxide slurry containing about 300 g Th/liter and 25 g U/liter in heavy water. A uranium oxide - plutonium oxide slurry is an important alternative fuel for the one-region reactor.

The solution-core reactor is the preferred two-region breeder because it offers the best possibility of obtaining the combination of high specific power and high conversion ratio that are required to achieve a doubling time of 10 to 15 years. Problems of fuel stability and corrosion and cooling of the reactor core tank may so limit the power that can be obtained from a single core as to make the solution-fuel reactors unattractive for large power reactors. The alternate to the solution-core breeder is the slurry-core, two-region breeder. Here the greater difficulty of separating fission products from the fuel and the low specific power required to limit the losses by absorption of neutrons in Pa-233 result in doubling times greater than 25 years. The optimum design based on economics has a conversion ratio only slightly greater than 1. Although the conversion ratios may be lower and the doubling times very long, the slurry-core, two-region breeder may have fewer limitations on size and power.

It is possible to obtain a conversion ratio greater than 1 in a one-region reactor by making the reactor very large. However, the one-region reactor is most interesting economically and technologically as a step in the development of the two-region, slurry-core breeder and as a low fuel cost power producer in which the deficit in conversion is made up by low cost U-235 in this era of plentiful fissionable material.

The 333-MWE (net) station that was chosen as a basis for comparison of the fluid fuel reactors is described here as a plant in which the heat is produced in solution-core, two-region breeder reactors. Changes which would result from the substitution of slurry-core, two-region and one- region reactors are described. The status of the technology for each system is presented and the important problems which remain to be solved are outlined. Finally, the costs of power are discussed and programs are outlined for the development of each system.

1. Two-Region Solution-Core Reactor Plant

A 333-MWE plant based on two-region solution-core power breeders would contain three reactors, each operated at a power of 380 MW thermal. Three reactors are used because the 2 g/liter concentration of fissionable material required in the core to breed effectively results in a small critical volume and the power density in the core

is limited by heat removal and corrosion considerations. The three reactors are operated separately but share buildings, control rooms, waste and ventilation, maintenance and many other facilities. Steam from all three reactors is piped to one 333-MW turbogenerator.

Each reactor vessel consists of a 4-ft-diameter by 12-ft-long cylindrical core operated at 320 MWT (75 KW/liter avg, 50 KW/l adjacent to core tank wall) surrounded by a 2-ft- thick blanket which is operated at 60 MWT (2.7 KW/liter). Under steady state conditions the core fluid is a D_2O solution 0.025 m in UO_2SO_4 (5 g U/liter at 275°C - 1.9 g U^{233}, 1.4 g U^{234}, 0.2 g U^{235} and 1.4 g U^{236}), 0.025 m in $CuSO_4$, 0.005 m in $NiSO_4$, and 0.025 m in D_2SO_4. Copper sulfate is present to recombine the radiolytic gas, nickel is a soluble corrosion product and the excess acid is added to stabilize the fuel. The blanket contains a 750 to 1000 g Th/liter slurry of thorium oxide in D_2O (8.5 to 11.4 volume percent solids). The steady state concentration of U-233 and Pa-233 in the thoria is 2.5 to 4.5 g/kg Th, depending on the blanket processing cycle. Molybdenum oxide or palladium catalysts, 0.02 m or 0.002 m respectively, are incorporated in the slurry to catalyze the recombination of radiolytic gas.

Fuel solution flows through the reactor core at a rate of 30,000 gpm, entering at 250°C (482°F) and leaving at 290°C (554°F). Fluid from the core divides into two parallel circuits where it passes through the steam generators and is returned to the core by 15,000-gpm canned-motor circulating pumps. In each circuit 170 MW of heat is extracted from the fuel to produce saturated steam at 400 psia and 445°F in the steam generators. D_2O is boiled at 325°C (617°F) to 335°C (635°F) in a pressurizer and surge chamber attached to the circulating system to control the pressure in the range 1750 to 2000 psia.

Under normal operating conditions 100 gpm of fuel at 290°C is passed through a hydroclone system to remove suspended fission and corrosion product precipitates and discharged through a counter current heat exchanger into a low-pressure system that operates at 15 to 100 psia. There heavy water is evaporated from the fuel at a rate of 10 gpm carrying with it the radiolytic gas, excess oxygen, fission product gases and iodine that were dissolved in the fuel. Iodine is stripped from the vapor by partial condensation of the D_2O and concentrated in a small reboiler; Xe^{135} is removed through the precursor I^{135}. Deuterium is recombined with part of the oxygen in a catalytic recombiner and the heavy water is separated from oxygen and fission product gases in a condenser. Fuel solution is returned directly from the low-pressure system to the high-pressure circulating system. Condensate is returned partly to the pressurizer and partly to the fuel circulating pumps as a purge for the bearings and rotor cavity. Oxygen and fission product gases are

passed through decay tanks after which most of the gas is recycled to the high-pressure system, but a part is discharged from the plant through cold traps, to remove small amounts of D_2O, and charcoal adsorber beds to delay the gases until 10-yr Kr^{85} is the only remaining radioactive gas. The gases are then discharged through a stack or the noble gases can be separated from the oxygen for further retention or sale. Common gas handling and disposal facilities are used by core and blanket systems of all three reactors.

Although the letdown and low-pressure system has been discussed above in connection with the removal of fission products and the production of condensate, it has several other intermittent functions. Fuel is added to the reactor through the low-pressure equipment. The fuel concentration, and by this means the operating temperature, of the reactor is regulated by control of the return of fuel and condensate to the high-pressure system. When the reactor is shut down, the fuel is contained in the storage tanks of the low-pressure system.

The thoria slurry which fills the reactor blanket is circulated through the reactor vessel at a rate of 15,000 gpm. It enters the vessel at 240°C (464°F) and leaves at 250°C (482°F). The flow rate through the blanket is determined primarily by the requirement to keep the slurry well mixed. The temperatures are kept low to help cool the core tank and to reduce the neutron losses. Because only 15 percent of the power is generated in the blanket the heat can be used efficiently at the lower temperature for heating the feed water before it enters the steam generators.

Slurry from the blanket is circulated through the heat exchanger (feed-water heater) and returned to the reactor vessel by a 15,000-gpm canned-motor circulating pump. There is no continuous circulation of slurry between the high-pressure system and a low-pressure system. The blanket is pressurized by heating slurry in the pressurizer which is attached to the slurry circulating system. Condensate is produced in the pressurizer to purge the bearings and rotor cavity of the pump. The blanket pressurizer is interconnected with the core pressurizer through pressure regulating valves to limit the difference in pressure across the core tank.

Although it is not in continuous use, a low-pressure system is provided for the blanket. It contains storage tanks for slurry and condensate, evaporators, recombiners, condensers, and feed pumps. The equipment is used primarily for storing slurry and adjusting concentrations, for feeding material to the reactor and for withdrawing material for processing. Gases released from the system are processed in the general plant facility.

To make the two-region solution-fuel reactor an efficient breeder, material is withdrawn from several places for processing for removal of fission products and fissionable material. Slurry is removed from the blanket low-pressure system for removal of fissionable material. Solids and some uranyl sulfate solution are removed from the hydroclone system for separation of corrosion and fission products. Condensate is removed from the iodine unit in the low-pressure system for separation of iodine and some fuel is removed from the low-pressure system for removal of soluble corrosion products, manganese and nickel. The processing steps are discussed in a later section.

2. Two-Region Slurry-Core Reactor

The upper limit on size of core which is critical with at least 2 g/liter if fissionable material is removed by use of a thorium oxide - uranium oxide slurry fuel. In practice, it appears necessary to have a slurry concentration of 100 to 300 g Th/liter so that fluctuations in slurry concentration which can be expected to occur will have little effect on reactivity. The corresponding critical concentrations of fissionable material in the reactor are 5 to 15 g/liter. Specific power (KW/g fissionable material) proposed for slurry fueled cores is lower than for solutions fueled cores to restrict the losses of neutrons by absorptions in Pu^{233}.

The 333-MWE power plant based on the two-region slurry-core reactor would contain one reactor operated at a power of 1140 MWT and one 333-MW turbogenerator. The reactor consists of a core 7 ft diameter by 21 ft long operated at 1060 MW (40 KW/liter) surrounded by a 2-ft-thick blanket which is operated at a power of 80 MW. Under steady state conditions a thoria, heavy water slurry containing 200 g Th and 14.3 g U per liter--6.9 U^{233}, 4.8 g U^{234}, 0.6 g U^{235} and 2.0 g U^{236}--is circulated through the core at a rate of 90,000 gpm. It enters at 256°C (493°F) and leaves at 300°C (572°F). Radiolytic gas is recombined in the slurry by a catalyst such as palladium.

Slurry from the reactor core is circulated at a rate of 15,000 gpm through six parallel steam generator and pumping circuits. Steam is produced in the steam generators at 400 psia and 445°F as with the solution-core reactor. The reactor is pressurized by heating slurry in a pressurizer attached to the circulating systems. Steam is condensed in the pressurizer to provide purge for the circulating pumps. No provision is made for removal of Xe^{135}.

The blanket slurry and high-pressure system are similar to those for the two-region solution-core reactor. Because this blanket is larger it appears necessary to circulate the 1000 g Th/liter slurry

at a rate of 30,000 gpm to obtain adequate mixing. Two parallel pumping circuits are provided but only one contains a heat exchanger; the 80 MW of blanket heat is used for feed-water heating. Core and blanket pressurizers are interconnected to prevent the pressure difference across the core tank from becoming excessive.

Although they are not in continuous use, low-pressure systems must be provided for core and blanket. They consist of storage tanks, recombiners and condensers, evaporators and feed pumps. Gas handling equipment is shared by core and blanket systems. Slurry is charged into the low-pressure systems for feed to the reactor and it is removed through the low-pressure systems for processing.

3. One-Region Slurry Reactor

The 333 MWE plant based on the one-region slurry reactor is nearly the same as that for the two-region slurry reactor except that the reactor vessel is a 12-ft-diameter sphere with a conical inlet and there are no blanket circuits. The reactor operates at 256 to 300°C and the high-pressure circulating system and the low-pressure systems are those described for the core system of the two-region slurry-core reactor.

D. Status of Technology

Development work both in and out-of-pile has brought the technology of fuels, materials and equipment for aqueous homogeneous reactors to the point where relatively little remains to be done to determine whether the solution core system is technologically feasible for large reactors. The technology of slurry fuels and equipment is not as well developed; there is only a small amount of experience in-pile. The number of critical questions for both solution and slurry systems which remain to be answered on the reactor experiment scale is small and the problems are well defined. Most of the equipment and maintenance methods for that scale of operation have been developed.

For the solution fuel it is necessary to explain the behavior of the fuel in the HRT and to determine from chemical, corrosion and radiation data, hydrodynamics experiments, and design studies whether a practical two-region reactor vessel can be built. Less work has been done with slurry fuels so a larger number of critical questions remain. Hydrodynamics experiments are required to determine the flow requirements of the reactor vessel. A satisfactory recombination catalyst must be demonstrated. It must be shown that the slurry materials retain their properties during very long circulation periods

and that long irradiation does not adversely affect the properties. Satisfactory solutions to the erosion and settling problems of slurries must be demonstrated.

Information concerning fuels and materials obtained on the reactor experiment scale can be applied directly to prototype and full-scale plants. A very large effort will be required to develop reactor and maintenance equipment for prototype and full-scale plants.

1. Two-Region Reactor Core Solution

The ideal fuel for an aqueous homogeneous reactor is a solution which has a low parasitic absorption cross section, is stable against precipitation of fuel constituents by thermal and radiation effects, is only slightly corrosive, and from which fission products and corrosion products can be separated easily. Dilute uranyl sulfate solutions most nearly satisfy these criteria but they fall short in a number of respects. Special care must be taken in the reactor design to compensate for the deficiencies.

Uranyl sulfate solution for the core of the two-region reactor contains 0.025 m UO_2SO_4, 0.025 m $CuSO_4$, 0.025 m D_2SO_4 in D_2O. Copper is present to recombine radiolytic gas, excess acid improves the stability and the solution must contain dissolved oxygen to prevent the reduction of uranium. The solution is thermally stable to 340°C. Above 340°C it separates into two liquid phases. One phase is 5 m or greater in concentration of uranium and contains approximately a proportionate amount of copper and nickel but slightly less than the stoichiometric quantity of sulfate. The second phase is depleted in metal ions but retains the acid. Both the concentration and the fraction of total metal ions in the heavy phase increase as the temperature of the two-phase mixture is raised. The heavy phase is thermally stable to temperatures above 500°C. Uranyl sulfate solutions are stable against radiation effects at least to the two-liquid-phase temperature. Very little is known about the stability of the heavy liquid phase in radiation at high temperatures and high power density.

During operation of the reactor introduction of fission and corrosion products alters the composition of the fuel and precipitates form at the reactor operating temperatures. Precipitation of fission and corrosion products makes possible the use of relatively simple hydraulic methods for separating them from the fuel so that this type of instability is of real concern only if it involves concentration of uranium or significant loss of copper catalyst. Iron, chromium, zirconium and similar elements hydrolyze and precipitate as oxides which may contain as much as 1 to 4 percent by

weight of adsorbed uranium. Barium, strontium, and most of the rare earth sulfates are only slightly soluble at the reactor temperature. Only a few of the products, e. g., cesium, rubidium, nickel and manganese, are very soluble and normally their concentrations are limited by processing so that the solution is stable against precipitation of uranium and copper salts at 300°C and lower temperatures. If the nickel concentration in the fuel is permitted to exceed about 0.005 m while the excess acid is maintained constant by acid additions, a basic copper sulfate precipitates until the nickel concentration is 0.017 m and the copper concentration is 0.018 m; then nickel sulfate precipitates. If the fuel is permitted to become depleted in acid to 0.015 m by copper and nickel, the system passes into a region of the phase diagram where a basic uranyl sulfate is the unstable phase. All of the products have retrograde coefficients of solubility so that instabilities appear at lower concentrations of nickel as the temperature is raised.

Recently it has been demonstrated that uranium and other soluble constituents of the fuel can be concentrated on heated surfaces by boiling if the velocity of fluid across the surface is low.

The effect of the fuel instabilities is to place several major requirements on design and operation of the reactor. All parts of the core tank in contact with the fuel must be kept below 300°C. Fuel in contact with the wall must flow at high velocity. Suspended solids must be removed rapidly and continuously from the reactor core and circulating system. The composition of the fuel must be controlled properly. Failure to comply with these requirements can result in damage to the reactor.

2. Materials in Uranyl Sulfate Solutions

Materials of construction for the solution core system are zirconium alloys for the reactor core tank, austenitic stainless steel--primarily type 347--for equipment external to the reactor core and titanium alloys for special applications in this external equipment.

Corrosion of zirconium alloys in-pile has been shown to increase with increasing temperature and power density--or fission rate--in fuel solution in contact with the metal. The corrosion rate at constant temperature and power density decreases with increasing uranium concentration, acidity and fluid velocity, factors which tend to reduce the relative effect of increased fissioning at the surface resulting from adsorbed or deposited uranium. The following are estimates of corrosion rates for Zircaloy in the reference fuel. They are based on the best correlation of data for in-pile loop and autoclave tests over a wide range of conditions.

Temperature °C	Solution Velocity ft/sec	Corrosion Rate, mils/yr at 25 KW/l*	50 KW/l*
250	5	11	11
	15	10	11
260	5	15	17
	15	13	15
280	5	27	32
	15	21	28
300	5	37	50
	15	27	40
330	5	53	85
	15	34	58

*Calculated maximum power density in solution adjacent to the core tank wall in reference reactor.

Stability could be improved and corrosion of Zircaloy reduced by additional acid in the fuel, but the acidity is limited to permit the use of stainless steel as a material of construction. Below 150°C stainless steel corrodes at a rate less than 1 mil/year. As the temperature is increased to 225°C the corrosion rate depends strongly on the concentrations of uranyl sulfate and acid; for the reference solution it is about 10 mils/year. At temperatures above 250°C, 0.1 to 1 mil of corrosion occurs while forming a protective film on the surface and the metal corrodes at a rate of a few hundredths of a mil per year. This situation holds at flow velocities up to a critical velocity above which rapid corrosion occurs for an unlimited time. The critical velocity decreases with increasing uranyl sulfate concentration and acidity and increases with increasing temperature to 300°C and above.

With the reference solution the minimum critical velocity is 15 to 20 ft/sec in standard, high turbulence tests at 250°C. It is about 55 ft/sec in smooth pipe. A velocity of 20 ft/sec is specified for reactor systems and care is taken in design to eliminate regions of abnormal turbulence. Pretreatment of the piping by exposure to oxygenated water is used to provide an additional safety factor. Experience in laboratory and in engineering scale tests has shown that a film which is formed by pretreatment protects the metal for thousands of hours under conditions where the critical velocity normally would be exceeded. Operation of the reactor under abnormal conditions for hundreds of hours should have little effect on the normal protective film.

Oxygen in the fuel prevents the reduction of uranyl ion and the subsequent hydrolysis and precipitation of the uranium which releases sulfuric acid and causes the solution to become very corrosive to stainless steel. The chloride content of the fuel must be kept below 5 ppm as a preventive against stress corrosion cracking. The oxygen concentration is normally kept in the range 100 - 500 ppm.

Special parts, such as pump impellers, heat exchangers which operate at 200 to 240°C or in which uranyl sulfate is boiled at reactor temperatures, special instruments and valve parts, which are outside the reactor core and are subjected continuously to unusually corrosive conditions are made of titanium alloys. These materials exhibit very low corrosion rates in uranyl sulfate solutions of all concentrations, show no velocity effects, and do not crack in the presence of chlorides.

Experience in loop tests both in and out of pile and in HRE-1 and HRE-2 indicates that austenitic stainless steels (AISI 347 in particular) are acceptable materials of construction for the reference fuel if proper precautions are used in design and operation of the equipment. Data from in-pile loop and reactor experiments lead to the conclusion that corrosion behavior of metals in equipment external to the reactor core is essentially the same as observed out of pile.

3. Slurry Fuels

Suspensions of thorium oxide, containing uranium oxide in solid solution, are the thorium-containing fuels for aqueous homogeneous reactors. No thorium solutions or sols have been found which satisfy the requirements of a core or blanket fuel. Problems encountered in the use of thoria slurries are related to the flow properties and maintenance of homogeneity of the suspensions, to recombination of radiolytic gas and to erosion by the thoria particles. Slurry fuels for the one-region reactor and for the core of the two-region reactor (200 to 300 g Th/liter) differ from the blanket slurry (1000 g Th/liter) to the extent that the concentration of the solids affects the properties.

Slurries which presently appear to be useful for a reactor are flocculated and behave as Bingham plastic or pseudo plastic materials. The yield stress and the coefficient of rigidity both increase with increasing volume fraction solids and decreasing particle size. A yield stress less than 0.1 lb/ft^2 and a coefficient of rigidity less than about 4 centipoises have been established as goals in the slurry development program so that the flow behavior will approach that of a Newtonian fluid and will not seriously affect the equipment design.

A wide variety of preparations satisfy these requirements in the 200 to 300 g/liter slurries. They are met in the 1000 g/liter slurries by use of particles of average size greater than about μ. Spheres have the optimum particle shape although properly sized cubes and platelets meet most of the requirements also. Dispersion of the particles by additives such as sodium silicate makes even the most concentrated slurries behave as Newtonian fluids but such slurries have not retained their properties indefinitely when circulated at high temperature or in radiation or the slurries have been difficult to resuspend when settled for long times.

Hindered settling rates for the promising slurries are 0.2 to 2 mm/sec at 30°C and increase to 1 to 3 cm/sec at high temperature. Slurries have been prepared which enter the compaction region of settling at concentrations from 800 to 3000 g/liter and settle to concentrations of 1200 to 4000 g/liter. Velocities in the range of 2 to 4 ft/sec are required to obtain fully developed turbulent flow in pipes and fully turbulent flow is required to maintain the slurries in uniform suspension.

In order for the slurries to retain their properties over long periods of time it is necessary that the particles retain their integrity and that the products which accumulate in fuel have an unimportant effect on the environment. Slurry particles produced by two methods have been found to retain their integrity adequately for test periods which vary from a few hundred to a few thousand hours. One method requires that the thoria be fired to 1600°C (or to temperatures as low as 1050°C for oxides which contain 8% U) to fuse the crystallites which form the particles. This material has a surface area of 1 m^2/g or less. The second requires controlled digestion of thorium oxalate crystals which are subsequently decomposed to thorium oxide and fired at 800°C. This material has a surface area of about 15 m^2/g.

The effect of adsorption of corrosion and fission products on the thoria particles has not been resolved. With hydrogen atmosphere the yield stress of slurry was observed to increase as corrosion products accumulated. Changing the atmosphere to oxygen resulted in a return to low yield stress, apparently as a result of change in oxidation state of iron and chromium adsorbed on the thoria. Little or no change in properties has been observed in most tests under oxygen atmosphere. Concentrations of fission products in most fuels should be considerably less than the concentrations of corrosion products. Blanket slurries have been irradiated in many tests for 150 to 300 hr and in one instance for several months at power densities of 0.5 to 5 KW/liter without obvious change in properties. Work was begun only recently on the irradiation of core slurries so no data are available.

Early in the slurry development program difficulty was experienced with "caking" -- the formation of hard deposits of thoria in circulating systems. This difficulty has been overcome by control of preparation and particle size in the development of particles which maintain their integrity.

One specification which has been placed on the slurry fuels is that they, like solution fuels, must contain a catalyst which will recombine radiolytic gas in liquid phase. Although this has complicated the development, two promising materials have been found. One is a molybdenum catalyst which is added to the slurry as molybdenum trioxide. This material appears to be satisfactory for the blanket slurry but has not worked well with slurries which contain more than 1% of uranium. The second catalyst is palladium deposited on thoria. This material works well both with the core and blanket slurries. Both catalysts are activated by radiolytic gas. The molybdenum oxide is activated with hydrogen and the palladium with hydrogen or oxygen before it is used to eliminate any induction period. Although both catalysts have been used to recombine gas in-pile in autoclave tests, considerable additional work is required before either one can be considered to be demonstrated for reactor applications.

Corrosion in slurry systems is essentially water corrosion with the added effect of erosion by the thoria particles. The combined corrosion-erosion rate depends on particle size and shape, and the environment. It increases in proportion to the concentration of solids and approximately the square of the velocity. If the particles are less than 1 μ, the shape is relatively unimportant; above about 5 μ, erosion can be severe unless the particles are spherical.

Severe erosive attack results from direct impingement of particles on objects in the line of flow and from the circulation of material in eddies in regions of flow separation. Corrosion-erosion rates with the reference slurry containing dissolved oxygen are less than 1 mil/yr for austenitic stainless steel piping systems where the velocity is 20 ft/sec or less and care is taken in the design to eliminate projections and areas of flow separation. Stress-corrosion cracking has been experienced in stagnant regions with oxygenated slurries which contained chloride, so strict control of chloride content is very important. Although it may be possible to alleviate stress-corrosion cracking by maintaining a large excess of dissolved hydrogen in the slurry the corrosion rates for stainless steel are at least four times those observed for oxygenated slurries.

Zircaloy and titanium have demonstrated excellent resistance to erosion by slurries, particularly oxygenated slurries. They are used for pump impellers and in other locations where the velocity is high, but where there is no severe throttling service.

Preliminary information concerning the effects of radiation on corrosion in slurry is being obtained from in-pile autoclave experiments. Indications are that the corrosion rate of Zircaloy is increased by radiation but that the effect is much less than encountered with solution fuels.

4. Reactor Equipment

With the exception of flow in the reactor core, the problems of reactor experiment scale equipment for the uranyl sulfate solution system have been solved and demonstrated in HRE-2. The flow problem is under investigation and there appear to be at least two satisfactory solutions for HRE-2. It remains to be determined whether these solutions to the problem will be satisfactory for large reactors. Less work has been done with slurries, but specification of the flow design for the reactor vessel and development of reliable valves and feed pumps appear to be the major remaining major equipment problems for HRE-2 scale slurry reactors. Work on larger equipment has been limited to preliminary testing of a 4000-gpm test loop by the PAR Project and design studies by both the PAR Project and the HRP at ORNL. However, much of the equipment for aqueous homogeneous reactors is adapted from equipment for pressurized water reactors to meet the special requirements of the uranyl sulfate and slurry fuels. The technology now being developed for large pressurized water plants will provide a firm basis for extension to large homogeneous reactor systems.

5. Primary System

a. Reactor Vessels

The reactor primary systems are the high-pressure circulating systems consisting of the reactor vessels, circulating pumps, heat exchangers, pressurizers and piping. The envisioned equipment can be constructed and assembled for all of the three reactors being considered here. Problems arise in ensuring that the equipment will have long life, that it can be maintained and that the reactor vessel can be operated satisfactorily at the design power.

A vessel for the two-region solution-core reactor has not been designed. The principles presented here have been suggested from HRE-2 data, in-pile loop tests and recognition of hydrodynamics problems. The vessel consists of a 4-ft-dia by 12-ft-long cylindrical zirconium-alloy core tank which is contained in an 8-ft-dia by 16-ft-long cylindrical pressure vessel fabricated of carbon steel clad with stainless steel. The pressure vessel is 6 or 7 in. thick; the core tank is 1/2 in. thick and is surrounded by, and integral with, a 3/8-in.-thick zirconium-alloy shroud which is used to direct

the flow of blanket fluid. Fluid enters and leaves the core through concentric pipes at one end. Cold liquid enters through the annulus with an axial velocity of 15 ft/sec and, along the wall, a tangential velocity of 15 ft/sec imparted by vanes at the inlet. It then flows the length of the tank, back through the center and out through the central outlet pipe. On the blanket side, the cold slurry enters the bottom of the pressure vessel, flows the length of the core through the annulus between the core tank and the shroud at a velocity of 15 ft/sec and then returns to an outlet at the bottom of the vessel through the large annulus between the shroud and the pressure vessel wall.

This design is proposed to provide high fluid velocity and low temperature required to obtain good cooling and moderate corrosion rates on the core tank. It is estimated that the maximum wall temperature can be kept below 260°C in the absence of transients and that the core tank life will be 10 to 20 years. The problem of transients has not been analyzed. Lower zirconium temperatures can be achieved at the expense of lowering the reactor temperatures or incorporating a separate 1000-gpm cooling circuit in a multiwall tank.

A vessel for the two-region slurry core reactor is similar but larger: 7-ft-dia by 21-ft-long core and 11-ft-dia by 25-ft-long pressure vessel. The 1/2-in.-thick core tank is constructed integral with a 1/2-in.-thick shroud and the pressure vessel wall is 8 to 9 in. thick. The flow pattern cannot be specified but there is some evidence that corrosion rates in slurry systems are considerably lower than in solution systems so the flow requirements may be less difficult.

The one-region reactor vessel is that proposed by the PAR Project but larger. It is a sphere with conical bottom inlet, 14 ft OD, 12-ft-dia core with a 6- to 7-in.-thick shell. The remaining space is occupied by thermal shields. Fluid enters the vessel as a jet at the apex of the inverted cone, traverses the length of the vessel, returns along the wall and leaves through an annulus around the inlet nozzle. Mixing in the vessel is promoted by induced internal recirculation.

None of the reactor vessels contain control elements. Dependence is placed on the negative temperature coefficient of reactivity to meet the regulation and safety requirements of the reactor.

Vessels of the sizes and shapes discussed above can be fabricated using present technology. Their feasibility for reactor application depends on development of satisfactory hydrodynamics and detailed mechanical designs. The details of bringing fluids into

the vessel, the structural details of the core tank, and the core tank to vessel connection are among the most important problems.

b. Heat Exchangers

Heat is removed from core and blanket fluids in shell-and-tube heat exchangers. High-pressure fluids are circulated through the tubes, feed water is heated in the shell of the blanket exchangers and low-pressure steam is generated in the shell of the core circuit exchangers. Designs incorporating straight or "U" tubes, arranged in horizontal or vertical bundles, with either fixed or floating heads have been investigated. All types propose the use of a recirculating saturated water leg with either an internal separating drum or a separate drum connected by risers and downcomers to the evaporating vessel. The exchangers are similar to those being developed for large pressurized-water reactor plants.

Principal problems associated with the primary heat exchangers arise from the size and the leak tightness requirements. The designs propose the use of carbon steel shells, heads and tubesheets with stainless steel cladding on surfaces in contact with the primary fluids and stainless steel tubes. The tubesheets are 4 ft. dia and 8 to 10 in. thick and each exchanger contains 3000 to 4000 tubes welded into the tubesheets. The large tubesheets are subject to high thermal stress. Special care must be taken in the design and fabrication to eliminate cracks, crevices and stagnation points, to have sufficiently smooth flow to prevent fretting corrosion and to provide for complete drainage of tubes and headers. These requirements may be difficult to meet in the complex configurations of heat exchange equipment. Absolute leak tightness is required to prevent the steam to the turbine from being contaminated and because the heat exchangers are, next to the reactor vessels, the most difficult items to maintain, they must remain leak tight for several years.

Large diameter high-pressure heat exchangers have been built to stringent initial leakage specifications; it remains to be demonstrated that the units will remain leaktight. Hold-up drums and monitoring of the steam are required to prevent radioactivity from reaching the turbine in the event of a leak. Designs with double tube sheets and dual tubes and intermediate heat exchange systems have been considered and discarded for economic reasons.

c. Fuel Circulating Pumps

Because of the stringent leakage requirements, canned-motor pumps are used exclusively for the fuel circulating pumps. Condensate is pumped into the rotor cavity to lubricate the bearings and prevent

the entrance of uranyl sulfate solution and slurry particles. The stators are heavily shielded to minimize radiation damage.

Basic designs are available for canned-motor pumps in sizes to 20,000 gpm. Modification of these designs is required to incorporate the top maintenance desired for homogeneous reactors and to provide other special features. Titanium or zirconium impellers, diffusers, thermal barriers, labyrinths and scroll liners may be required for both solution and slurry pumps. Experience indicates that long service life can be expected from solution fuel pumps. Difficulties have been experienced with erosion of impellers, seal rings and scroll liners in slurry pumps. Recent experience with Zircaloy parts in small pumps has encouraged the belief that long life can be obtained from slurry pumps but they will require more frequent maintenance than the solution pumps.

d. Pressurizers

Pressurizers proposed for the reference plants are essentially enlarged units that have been used on HRE-2 and on the slurry development loops. The solution system pressurizer involves a surge drum which is connected by means of large-diameter piping to the core circulating circuits and an attached electric boiler. Condensate produced in the low-pressure system is pumped to the boiler and evaporated at rates as high as 10 lb/min to provide the desired steam pressure.

Slurry pressurizers are tall columns through which a part of the slurry is circulated continuously. The column acts as a settling chamber leaving clear water at the top. The water is heated to provide the desired vapor pressure and a part of the vapor is condensed to provide a purge of about 10 lb/min to the rotor cavity of the slurry circulating pump.

Similar, but much smaller, pressurizers have been operated satisfactorily for both solution and slurry fuels. Problems may be encountered in providing satisfactory level control, heating and connections to the main circulating systems in large units.

e. Piping and Valves

Piping components for 2000 psi service are available in AISI/300 series stainless steels in all the sizes required for these reactors. All welded piping systems are preferred but the equipment can be assembled with flanged connections and leak detection equipment as in HRE-2 if this simplifies the maintenance of equipment. Ring joint flanges have proved satisfactory in sizes to 4 in. at 2000 psi in

HRE-2 and only the main circulating lines are larger in the high-pressure system. The feasibility of using a few large flanged joints requires a demonstration that 18-in., 1500-psi flanges can be kept tight through many thermal cycles and when subjected to high bending moments.

Valves in most of the sizes required for the solution-core system have been demonstrated in HRE-2. The largest required for solution or slurry are 4-in. valves for the dump lines and they must be developed. Work on valves for slurry systems is in progress; erosiveness of the slurry makes the problem more difficult. Zircaloy and tungsten carbide show promise of satisfactory life as trim materials for some applications but no satisfactory material has yet been found for severe throttling service.

6. Low-Pressure Systems

Low-pressure equipment is provided for storing fuels when the reactor is not operating, for adjusting the fuel concentration during startup and during normal operation, for charging new fuel to the reactor and discharging fuel and fission product gases to the processing facilities, and for providing clean water for washing the high-pressure system prior to maintenance operations. Equipment required to carry out these operations consists of storage tanks, evaporators having a total capacity of about 300 lb/min for the 1140-MWT station, recombiners, condensers and feed pumps. The low-pressure systems are designed for 300 psia service to contain the reactor fluids if they are discharged without cooling in the high-pressure system. Normally they are operated at 15 to 100 psia.

The equipment is operated intermittently for slurry systems but is used continuously with solution fuels to remove iodine and thereby the Xe^{135}. This requires an additional partial condenser and reboiler with a capacity of about 10 lb/min and the feed pump must have a capacity of 1000 lb/min instead of 100 to 200 lb/min. There is no change in size of compressor used to recirculate oxygen to the reactor but it must operate continuously.

The operability and reliability of most of the low-pressure equipment for solution fuels has been established on small scale in HRE-2. Exceptions are the exact counterparts of the 1000 lb/min feed pump, the gas compressor and the iodine removal unit. Larger multistage canned-motor pumps will have to be developed for the reactor. A single-stage turbine pump having a capacity of 100 lb/min against 300 psi head has performed satisfactorily in 5000 hours of testing. A three-stage, 2-cfm, diaphram type gas compressor, sufficiently large and designed for this application, is being installed

in a test facility at the present time. Silver-plated stainless steel mesh is used for removing iodine in HRE-2. Adsorption in liquid phase is considered to be a more satisfactory method for the larger reactors.

Components for low-pressure slurry systems have been designed and tested but have not been operated in integrated systems even on small scale. It has been demonstrated that the slurries can be kept suspended or can be resuspended after settling in several types of tanks by use of steam sparging, pumping and mechanical stirring. Evaporators, settling tanks, hydroclones and centrifuges have been used for concentrating slurries. The major foreseeable problem is the feed pump. Turbine pumps cannot be used in slurry service. Trim materials have not been found that will last more than 1000 hr in check valves of positive displacement pumps. Batch charging methods in which slurry is pressurized in a container to the circulating system pressure before valves in the connecting lines are opened have been used successfully in loop experiments. This method may be acceptable for larger systems since the charging and discharging are normally intermittent.

An important problem common to the low-pressure systems is that of inventory control. Facilities must be provided for sampling contents of the tanks and for providing continuous indication of the weight or density and volume of materials in all tanks. Inventory determination by weight and sampling as used in HRE-2 is the most satisfactory method that has been developed but many improvements will be required to obtain accurate control in a large plant.

The off-gas system is an enlarged version of one which has performed satisfactorily on HRE-2. Gases from the low-pressure systems pass through cooled tanks where the short-lived activities decay. Part of the gas is recycled to the reactor from this point; the remainder is routed through cold traps or alumina dryers to remove traces of D_2O and are discharged into carbon beds to delay the gas until all but the 10-yr Kr^{85} has decayed. Then it is discharged through a stack or processed for removal and bottling of noble gases if this is required. The carbon beds are water cooled to 150°F. Although it is possible for the beds to ignite since they contain oxygen, calculations, laboratory tests and experience with HRE-2 show that a fire can be stopped and the activity retained by closing valves at the inlet and exit of the affected unit. The bed can be used again if it is isolated until the gaseous activities decay and then is purged or evacuated to remove adsorbed carbon dioxide. Molecular sieve materials can be used as adsorbents in place of carbon but they are less efficient.

7. Reactor Plant Instrumentation

Instrumentation and control systems for aqueous homogeneous reactors are similar to those used in modern high-pressure steam power and chemical plants. Some nuclear instruments are required but the largest part is direct process instrumentation. Problems attendant to the special requirements of fluid fuel reactors has required development of special components. Special liquid-level transmitters, level probes, differential pressure cells, weigh systems and flow transmitters were developed for use in HRE-2. In most instances they have proved satisfactory and provide a basis for improved devices for large solution fuel reactors. For slurry systems, substantial improvements must be made to instruments now in use for measuring flow, density level and pressure before they can be used in a power reactor.

A combined electric-pneumatic signal transmission system is used in HRE-2 and would be recommended for large reactors. In the control room, electric signals from primary variable sensing elements are converted by transducers to pneumatic signals and these are used to actuate miniature pneumatic display instruments and pneumatic valves in the reactor. The ability to use all metal, radiation resistant construction for transmitters and valve actuators makes the pneumatic system particularly attractive for application in high radiation fields. Radiation damage to the primary electrical elements is avoided by the use of inorganic electrical insulators such as ceramic, mica, magnesium oxide and magnesium silicate.

8. Turbogenerator Plant

The plant which utilizes steam from the reactor differs from a fossil-fuel plant in two respects. The steam is low-pressure saturated steam and protection must be provided to prevent the turbine equipment from becoming highly contaminated in the event of a burst tube in a steam generator. Shielded drums are provided to delay the steam for 5 sec between the steam generators and the turbine. The steam is monitored and the drums are isolated by valves in the steam line if activity is detected. The turbine is the same as those being designed for pressurized water and boiling water reactors. They are presently listed in sizes to 300 MW.

9. Reactor Building

The power plant contains a variety of structures, the largest of which are the reactor and turbogenerator buildings. Others are provided for fuel processing, waste handling, maintenance, water treatment, etc. The reactor building is considered here because of its

special requirements. The turbogenerator building is conventional and should require no discussion. Where other structures are unique they will be described in conjunction with their functions.

Two types of buildings have been considered for housing aqueous homogeneous reactors. They differ primarily in the philosophy of containment and it is difficult to decide which will be required for large plants or whether the two will be combined. The first was used in HRE-2 and involves installing the reactor and auxiliaries in compact cells which consist of an inner metal liner, the complete biological shield and an outer metal shell assembled as an integral structure. The top of the cell is largely removable in sections which are keyed into girders that support the shield blocks in two layers. A metal diaphragm is installed between layers and welded to the girders so that the cell is sealed completely during operation. Service and instrument lines which penetrate the cell walls terminate in service galleries adjacent to the cells. Cells and operating galleries for one or more reactors are contained in a canyon which communicates with the maintenance facilities. A crane with shielded cab is provided for use in maintenance operations and for transporting equipment from the cells to the maintenance storage pool.

The building to house three solution-core reactors is 300 ft long by 100 ft wide, divided into three reactor equipment cells and adjoining service galleries. The cells and galleries are 60 ft deep, and the crane bay rises to a height of 50 ft above the cells. The building for the two-region slurry core or the one-region slurry reactors would be about 125 ft wide by 225 ft long.

The second type of containment is that proposed by the PAR Project. It involves constructing cells similar to those described above for the reactor and auxiliaries except that the top is not sealed and the shielding in most areas is less than that required to reach biological tolerance when the reactor is operating. These cells and the service galleries are housed in a large spherical or cylindrical steel vessel which provides the final containment in the event of the maximum credible accident. A crane with shielded cab is installed in the containment vessel for maintenance operations, and provision is made for removing equipment to the shops. A 2-ft-thick concrete wall surrounding the containment vessel completes the biological shielding during operation and reduces the radiation levels in the surrounding areas in the event that fission products are dispersed in the sphere. Three 150-ft dia spheres or three vertical cylinders approximately 120 ft dia by 180 ft would be required for the two-region solution-core reactors. The two-region-slurry core or one-region slurry reactor could be contained in a sphere about 200 ft dia.

Both types of containment would retain completely the radioactivity from small leaks and spills or the most likely major spill, rupture of a small pipeline that had become weakened by erosion or corrosion. The more hazardous, though far less likely, occurrence would be the "maximum credible accident," rupture of the large primary piping or the reactor vessel itself.

In the event of a major rupture, the pressure in the completely sealed cells, which are normally kept at 7 to 10 psia, would rise to a maximum of about 50 psia and then gradually decline to less than 15 psia in 8 hr. A fair assumption as to dispersal of activity would include all of the gaseous component and 30 percent of the activity associated with the solid phase. Ideally these would be retained within the cell; however, with the large number of penetrations, a complete seal is practically impossible. A reasonable leakage rate is estimated to be about 1% per day when the pressure is 50 psia, so during the first 8-hr period approximately 0.1% of the dispersed activity could be expected to leak from the cell to the service galleries. Ventilation from the galleries and the crane bay is passed through filters and washed so that the activity discharged from the stack is limited to krypton and xenon isotopes. By proper control of ventilation in the galleries the contaminated areas can be kept to a minimum.

In the event of such a major rupture in the large vapor container, the seals of the primary concrete cell would relieve to the vapor container. The pressure in the vapor container would rise to a maximum of 12 psig and gradually decline to about 6 psig after two hours; airborne activity would be uniformly dispersed throughout the cells and the space within the vapor container.

Three elements of radiation hazard following such an unlikely accident are of concern: direct radiation through the secondary concrete shield, "sky-shine" or scattered radiation from the open top of the concrete shield, and airborne contamination from small leaks or penetrations through the vapor container. In the PAR studies* it was calculated that the direct radiation level at a point on the ground 50 ft from the concrete cylinder will initially be 100 r/hr, decreasing at such a rate that the time required to accumulate a dose of 25 r will be 45 minutes. The dose rate at the end of a month, assuming no decontamination in the meantime, would be 1 r/hr. At a point 1000 ft from the base of the reactor, the

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* The numbers apply to a 450-MWT, one-region slurry reactor and are based on all the fission products becoming airborne. They would be increased in proportion to the total reactor power and decreased in proportion to the fraction assumed to be dispersed to apply to other reactor systems.

direct radiation dose rates will be 5 r/hr and 50 mr/hr, initially and at the end of one month, respectively. At a point 2500 ft from the reactor, assumed to be the site boundary, the corresponding two rates will be 100 mr/hr and 1 mr/hr, respectively.

The results of a rough calculation indicate that the effect of sky-shine at the 50-ft point will be about equal to that of direct radiation; thus, based on a limiting dose of 25 r, plant personnel would have about 20 minutes to evacuate to the control room, which has additional local shielding. The effect of sky-shine at points more remote is significantly less important.

It is assumed that at 12 psig, the vapor container will leak at a daily rate of 0.1% of its free volume. The resulting dose rate from the radioactive plume, at a point 2500 ft downwind of the reactor under the most unfavorable meteorological condition of "fumigation," will be 4 r/hr after 15 minutes and 2 r/hr after 15 hours. A more favorable wind condition of "coning" would halve those rates.

As to breathing tolerances, the most hazardous element released would be radioiodine. Under the extreme assumption that all of the iodine in the reactor is uniformly dispersed, the maximum downwind concentration 2500 ft from the reactor is approximately 3×10^{-6} μc/ml, or about double the maximum permissible concentration for an 8-hr exposure. The calculation does not take credit, however, for the tendency of iodine to nucleate on airborne particles and thus to be washed down to the bottom of the vapor container.

The most desirable type of construction is one which confines the contamination to a minimum area and from this standpoint the sealed cells are preferred. Feasibility of this system depends on the ability of the builders to ensure freedom from leaks through the penetrations. In the event of a catastrophic rupture of the primary system in a plant built with unsealed cells in a large vapor container, contamination is wide-spread in the container but plant personnel will have adequate time to move to the shielded control room and there will also be time for an orderly evacuation of people living in the immediate neighborhood of the plant site.

10. Reactor Operation and Control

Aqueous homogeneous reactors rely for control completely on variable fuel concentration and temperature coefficient of reactivity rather than poison-containing control rods. Shim control to compensate for fission product poisons, to change the operating temperature, to put the reactor into operation and to shut down is accomplished by adjusting the core fuel concentration.

In starting the reactors the core fuel is concentrated in the low-pressure system and heavy water is pumped into the reactor core and core circulating systems. At the same time the slurry is pumped into the blanket of a two-region reactor. Circulation of the fluids is started when the filling is completed, pressurizing is started and steam from an auxiliary supply is admitted to the heat exchangers. The reactor systems are heated at a rate of 40°C (104°F) per hour until the temperature reaches about 200°C (392°F). During this time fuel is pumped into the core system but at a rate that will keep the reactor subcritical. Fuel is added at that temperature until the reactor becomes critical and then is continued so that the temperature rises to 250°C in about two additional hours. Release of steam from the steam generators to the turbine is begun at 250°C and the steam release rate and addition of fuel to the reactor are controlled to keep the reactor inlet temperature at 250°C as the reactor outlet temperature and power are raised to the normal operating level.

In the case of the solution-core reactor the addition of fuel is stopped and only small concentration changes are made to compensate for the growth of small amounts of poison or to change the temperature level. Feed to the slurry-core reactors is continued as the Xe^{135} builds in and further until the temperature begins to decrease slightly as a result of increasing resonance absorption. At this point fluctuations in concentrations will have little effect on the reactivity and the startup is complete.

The temperature rise of the fluid passing through the reactor is 40°C so rapid changes in power output can be made without serious thermal shock to the reactor equipment. Changes in output are effected either by regulating the flow of steam at the turbine or by regulating the flow of feed water to steam generators to maintain a constant steam pressure. A normal reactor shutdown is accomplished by reducing the fuel concentration in the core by discharging fuel to the low-pressure system and returning condensate to the high-pressure system. During this several-hour cooling period the concentration is reduced at about the maximum capability of the low-pressure system and the flow of steam from the steam generators is regulated to control the temperatures to prevent the thermal stresses from exceeding safe limits. When the equipment has cooled to about 100°C, the fluids are discharged to the storage if maintenance is to be done or they are retained in the reactor if the shutdown is for other reasons. If a serious leak develops in the equipment, an emergency dump can be made from high temperature and pressure in 30 min.

Dependence is placed on the negative temperature coefficient of reactivity to meet the regulation and safety requirements of the reactor. The temperature coefficient varies from $-3 \times 10^{-3} \Delta k/k$ per °C for the two-region reactors to $-1 \times 10^{-3} \Delta k/k$ per °C for the large

one-region reactors. As there is no requirement that the short-time variations in power be kept below several percent, regulation is achieved with ease. Control of fuel feed rate, steam release rate from the steam generators and pump startup rate is provided to limit the foreseeable reactivity rise rate to values that can be accommodated easily by the system.

Mechanical operation of the plant is a combination of standard power plant operation and radiochemical plant operation. Operating difficulties which differ from those in the normal power plant are concerned with control of inventory and transfer of radioactive fluids. Because the fuel is fluid, care must be taken and controls provided to prevent the inadvertent escape of radioactive material. The major hazard in the operations is the taking and transporting of fuel samples, a part of the inventory procedure. With continuing development, equipment will be devised so that only minor hazards exist there.

11. Maintenance of Reactor and Other Radioactive Equipment

A major problem of fluid fuel systems and one that may be a determing factor in their ultimate economic feasibility is that of maintenance. Because the systems are highly radioactive the maintenance must be done remotely or semi-remotely. Special tools are required, the operations are time consuming and care must be taken to control contamination to prevent its spread throughout the plant. Maintenance of radioactive equipment on a large scale has been demonstrated in the processing plants at Hanford and Savannah River. Maintenance of aqueous homogeneous reactors has been demonstrated on pilot plant scale with HRE-2. It remains to be determined whether maintenance of a large power reactor plant can be done economically.

The maintenance methods chosen have a profound influence on major elements of plant design. In the first place, they dictate the arrangement and spacing of almost all components and runs of piping, and as a consequence the physical size of the plant as a whole. Second, it is found that all components, including those that are quite conventional, must be specifically designed with a remote maintenance procedure foremost in mind.

In any plant, components may be either repaired in place during shutdown or, alternatively, replaced by a spare and repaired after the plant is restarted. The choice depends in each case on the relative costs of downtime against the cost of the spare component. In a fluid reactor plant, this is no less true than in a conventional plant, except that the difficulties of remote in-place repair and the

high cost of downtime keep to a very small number the components which will be repaired in place.

In HRE-2 some maintenance operations are done with special tools through special shields with the cell dry. However, operations which involve removal of large pieces or opening of large areas of the cell are done by use of long handled tools with the cell flooded to provide shielding. Although it is possible to design a large plant for the same type of combined dry and underwater maintenance, the PAR Project chose remote dry maintenance for all operations on a large reactor. This general concept would be used in the plants proposed here except that the cells would be designed so that they could be flooded to make semi-direct work possible if necessary.

In the PAR design, the basic maintenance tool is a shielded-cab bridge crane which has access from above to the majority of the components. The functions of this crane are to lift the concrete slabs which form the roofs of the several large compartments; to place positioners, cutters, welders, wrenches and other remotely-operated tools; and to remove the affected component to the maintenance pool adjoining the reactor building. The crane is assisted in these operations by bridge- and wall-mounted manipulators which are placed at strategic locations within the compartments. Wherever advisable, small components are grouped for removal as a sub-assembly and in each case the lifting point is directly over the center of gravity of the component or sub-system.

An important fraction of the PAR effort was devoted to the development or adaptation of the special machines needed for the execution of the concept described above. At the time of suspension of the Project, these were at different stages of design or development, but in all cases encouraging progress had been made. An experimental model of a 10-in. automatic welding machine applicable to remote operation was built, and a number of full-strength welds were made in stainless steel pipe. A combination remote positioner-cutter-welder for one-inch pipe and its control console was built and tests of it were about to begin. Preliminary designs were completed for a 16-in. floor-mounted positioner and for a pipe-mounted positioner of intermediate size. Progress had been made in the adaptation of ultrasonic techniques for weld inspection, but a considerable amount of work remained to demonstrate its feasibility.

Flanges may be used in many of the pipe lines in the low-pressure system, in small lines in the high-pressure system, and in the largest lines of the primary system if remote welding proves to be unsatisfactory or uneconomical. Ring joint flanges could be used in all sizes, but equipment would have to be developed for remote positioning and bolting of the flanges and for machining damaged grooves. If such

operations appear to be feasible, the leak detection system, which has been demonstrated on HRE-2, provides an effective means for testing the flanged connections.

There are two major exceptions to the applicability of the techniques and machines described above. Repair of a reactor vessel would require that it be removed to the maintenance pool. Depending on the damage, the vessel might be discarded or it might be repaired by semi-remote methods using tools prepared for the particular job.

Special provisions must also be made for repairing the steam generators. In the PAR work methods were under development for doing the steam generator maintenance in place from rooms adjacent to each of the steam generators. Design and some development were done on equipment for surveying the tube sheets, detecting leaky tubes, cutting off the tube ends, plugging the tubes and welding the plugs. Successful tests were made of the plugging and welding elements of the machine.

At the present time, no estimate can be made as to the frequency with which component repairs will have to be made. Estimates have been made as to the length of plant down-time which might be incurred by some of the major maintenance operations. By way of example PAR Project estimated that it would take about 5 days from full-power to full-power, to remove and replace a primary pump. Included in this time are 24 hours for a normal plant shut-down and for ventilation prior to opening the primary compartment, and 32 hr for plant start-up after the compartment is closed. To open a steam generator, search out and repair one or more tube leaks, and reclose the generator would require about 6 days, exclusive of the time required for plant shutdown and startup. Three days would be used for the leak-detection survey itself and 2 days for the rewelding of the two 16-in. access nozzles.

A maintenance facility is provided adjacent to the reactor building for repairing equipment and for preparing for disposal equipment or parts that must be discarded. The facility is provided with decontamination equipment for removing surface activity and a storage pool 18 ft by 40 ft by 30 ft deep capable of storing heat exchangers or reactor vessels.

A portion of one end of the pool projects into a hot-shop, wherein items which have been allowed to decay in the pool but which are still too radioactive for direct maintenance can be reclaimed. The hot-shop is approximately 20 ft wide, 60 ft long and 20 ft high, surrounded by 5-ft-thick concrete walls for biological shielding. Since all operations are performed remotely, the shop is equipped with maintenance devices which are operable from outside of the shield.

Such devices are a 10-ton overhead crane with an auxiliary hook for light equipment, a heavy-duty overhead manipulator, and wall-mounted manipulators for light and intermediate service. In order to observe and direct the operations, the shielding walls are penetrated at appropriate places with windows, and at two of these observation points small master-slave manipulators are mounted to facilitate maintenance of a more delicate nature than could be handled by the heavy-duty manipulators. A turntable is provided to facilitate work on large heavy items.

In addition to the hot-shop the maintenance facility contains the mockup shop with the jigs and fixtures that are required for fabricating new piping and equipment for the reactor. Also there are the instrument shops, machine shop and general maintenance and stores facilities for the entire plant.

12. Waste Disposal Systems

Wastes from the reactor plant consist of the ventilating air, solids which accumulate from maintenance and laboratory operations and from ion exchange beds and liquid wastes from spills, sampling and analytical operations, regeneration of ion exchange beds, equipment decontamination, and the fuel transfer areas where carriers are charged for shipment to central processing facilities. Additional tankage required for handling wastes from an on site processing plant would be included in the waste area but the amount is not specified here and the cost is included as a cost in the operation of the processing plant.

Ventilating air withdrawn from the plant facilities at a rate of about 50,000 cfm is washed in a spray chamber, filtered, and discharged up a 250-ft stack. Solid wastes are stored in a burial ground; fine residues or large numbers of small pieces are combined in drums before burial.

More elaborate facilities are required for the liquid wastes. Spills or laboratory wastes that involve a considerable quantity of fissionable material are discharged to a 2500-gal critically safe storage tank. Waste is discharged from this tank to an evaporator where it is concentrated, and then transferred to the on site processing plant or to the fuel carrier charging area for shipment.

Wastes that do not involve large quantities of fissionable material are discharged to one of four 60,000-gal storage tanks. From the storage tanks they are sent to evaporators for concentration. Condensate from the evaporators is collected in one of two 10,000-gal condensate tanks, sampled, discharged to the ion exchange units as make-up for

the maintenance pool or to a low level storage pond, or recycled if the activity level is too high. Bottoms from the evaporators are discharged to a 200,000-gal permanent storage tank or, if sufficient uranium is involved, routed to the carrier loading area.

E. Fuel Processing

The processing of aqueous homogeneous reactor fuels is carried out to remove fission products and corrosion products which absorb neutrons and cause solution fuel instabilities. For the blanket of a two-region reactor, processing has the primary purpose of separating fissionable and fertile material, and removal of fission and corrosion products is of secondary importance. The Thorex process is readily adaptable for the recovery of decontaminated U-233 and thorium from both solution fuel and slurries for aqueous homogeneous reactors.

1. Solution Fuel Processing

Uranyl sulphate solution from the reactor core is processed through a system of hydroclones at a rate of 100 gpm, one reactor volume per hour, to concentrate insoluble fission and corrosion products into a small volume of fuel solution in the underflow tank. Slurry discharged from this tank at convenient intervals represents the major withdrawal of material from the fuel system. The use of hydroclones for this purpose was demonstrated on HRE-2, where a single 0.5-in. hydroclone removed the insoluble materials. The demonstration was not completely satisfactory because only 10 to 20% of the solids produced in the reactor reached the hydroclone.

The fuel solution after passing through the hydroclones, is let down to the low-pressure system, flashing about 10% of the D_2O. This D_2O vapor effectively strips iodine, xenon, krypton, and dissolved O_2 from the fuel solution as demonstrated with HRE-2. Separation of these gases from the D_2O can readily be effected in a liquid-gas absorption tower. Iodine is relatively non-volatile in pure water at the lower temperature and is removed from the bottom of the tower dissolved in a small stream of D_2O. The bulk of the D_2O, sufficiently pure for reuse without further treatment, may be recovered by condensing the overhead from the tower, and the xenon, krypton, and O_2 are discharged as non-condensables.

With hydroclone processing and continuous removal of iodine and rare gases, soluble nickel from corrosion of stainless steel becomes the major chemical and nuclear contaminant in the fuel solution. The concentration is kept at 0.02 m, an acceptable level from the standpoint of neutron poisoning, by the withdrawal of fuel

solution from the hydroclone underflow tank at the rate of 60 liters/day. However, since dissolved nickel contributes so heavily to solution instability, it is advisable to remove nickel at a rate sufficient to hold the concentration to 0.005 m or less. It is proposed to treat about 200 liters/day of fuel solution from the reactor low-pressure system in an electrolytic cell for removal of copper and nickel. Such a cell works best with a mercury cathode. After adding fresh copper catalyst, the treated fuel solution can be returned directly to the reactor. The mercury can be purified for reuse by passing sulfuric acid solution through the electrolytic cell and reversing the electrode potential. The resulting nickel, copper, sulphate solution becomes a radioactive waste. This method has been developed and successfully tested only in the laboratory.

The electolytic removal of nickel is highly desirable for a reactor operated as a breeder where the amount of fissionable material in the chemical plant becomes an important consideration in determining doubling time, because little decay time is required before electrolysis. However, in case doubling time is not an overriding consideration, sizing the hydroclone underflow withdrawal rate to control nickel appears to be the simplest, most economical procedure.

The hydroclone underflow tank contents and the iodine-D_2O solution from the iodine removal system are combined with slurry removed from the reactor blanket low-pressure system for subsequent D_2O and U-233 recovery as described in the next section.

2. Slurry Blanket Processing

Slurry for processing is removed from the blanket system at convenient intervals. The total amount of slurry processed for a 333-MWE station varies from about 1350 liters/day (1350 kg Th/day) for a 3 reactor power breeder station operated to obtain a 6- to 10-yr doubling time, to only about 120 liters/day (120 kg Th/day) for the blanket of a slurry fuel reactor operated to minimize fuel cycle cost. The slurry, combined with the underflow from the hydroclone, is then sent to D_2O recovery. The D_2O is recovered for immediate reuse by evaporating the slurry to dryness and heating the solids to about 300°C. This will drive off 99.9% of the D_2O, based on laboratory tests. Larger scale tests indicated that foaming during evaporation and entrainment of ThO_2 particles in the off-gas are the most serious problems associated with D_2O recovery.

After D_2O recovery, the material is stored for about 150 days for decay of short-lived fission products, Pa-233 and Th-234, before

solvent extraction processing. This period is sufficient to reduce the U-233 loss as Pa-233 and the Th-234 contained in the thorium to insignificant levels. If processing is done on the reactor site, the solids may be dissolved in HNO_3 following the drying step and stored as a solution for decay. If the material is to be shipped to a central chemical processing plant, both the drying for D_2O recovery and decay storage may be done in a suitably designed shipping container.

By using two full cycles of Thorex solvent extraction, the thorium and U-233 are recovered essentially free of fission product activity, but both contain isotopic contaminants that make them difficult handling problems. The uranium contains U-232 which adds to the alpha hazard and whose daughters are energetic gamma emitters. The thorium contains appreciable Th-228 after several cycles through the reactor. This also contributes added alpha activity and energetic gamma emitting daughters.

3. Slurry Fuel Processing

In the case of a slurry fuel essentially all the fission products and corrosion products are adsorbed on the surface of the particles. There is a possibility that I_2, Xe and Kr can be stripped from the slurry in the reactor, but the technology for accomplishing this does not now exist. Therefore, at present, slurry fuel processing must be considered as removing the slurry from the reactor, recovering D_2O by evaporation, decay storage, and decontamination by Thorex. For a 333-MWE installation the amount of slurry fuel processed per day is about 200 liters or 50 kg of thorium.

Laboratory tests have indicated that it may be possible to remove the bulk of the fission product and corrosion products from the slurry with a HNO_3-H_2O_2 leach, without dissolving the slurry. However, more work is necessary to determine whether or not this represents an improvement over sending all the thorium through Thorex and reconstituting the slurry from decontaminated thorium and uranium.

4. Reconstitution of Fuel Solution

This has been satisfactorily accomplished in two ways, both equally attractive. First, uranium from the Thorex product uranyl nitrate solution can be sorbed on a cation exchange resin, eluted with H_2SO_4, the UO_2SO_4 evaporated to dryness and redissolved in D_2O. Second, the uranium can be precipitated from the uranyl nitrate solution as UO_4, fired to anhydrous UO_3, and redissolved in D_2SO_4 solution.

Both methods can be carried out rapidly enough to avoid serious radiation problems from the daughters of U-233

5. Reconstitution of Blanket Slurry

The preparation of pure ThO_2 for the blanket of a two-region reactor involves precipitating thorium oxalate from the Thorex thorium nitrate product solution and decomposing this material to the anhydrous oxide by firing. Present indications are that a final firing temperature of 800-900°C is sufficient if care is taken to thoroughly digest the thorium oxalate after precipitation.

The particle size of the final oxide is controlled by the conditions under which the oxalate is precipitated. The concentration of the reagents, the rate of reagent addition, and temperature all affect particle size. Present practice consists of adding 1 M oxalic acid to 1 M $Th(NO_3)_4$ at 25°C and at a rate determined to produce a final average particle size of 2.2 ± 0.2 microns. The oxalate precipitate is then digested for 48 hr at 85°C, filtered and fired to 800°C. The present scale of this operation is about 200 lb of ThO_2/batch.

If subsequent work indicates that a firing temperature well above 1000°C is required, the procedure becomes more complicated. The oxide, after being fired in the electric furnace, must be transferred to a gas fired furnace, fired to the desired temperature, discharged, slurried in water containing oxalic acid to disperse the oxide, classified to remove oversize sinters, centrifuged, and refired in an alectric furnace to remove the oxalic acid.

6. Preparation of Fuel Slurry

The simplest, most satisfactory method of adding uranium to controlled particle size ThO_2 is the following. The thorium oxalate is prepared as before to control the final particle size; however, the thorium oxalate is fired only to 650°C to produce a high surface area oxide. Urania is deposited on the preformed ThO_2 particles by slurrying the thoria in a solution of uranyl ammonium carbonate which decomposes to UO_3 on heating to 100°C. The UO_3 precipitates on the ThO_2 particles, and the solids are recovered by centrifugation. The uranium is then diffused into the ThO_2 by firing at 1050°C to give a homogeneous mixed oxide. This oxide has been produced in 200-lb batches.

The decay products of U-232 and Th-228 will make fuel slurry reconstitution a highly radioactive operation. Shielding will be

required and all materials must be carefully contained to prevent spread of the alpha active materials.

F. Economic Appraisal (Project Director's Appraisal)

Power costs are composed of fixed charges on invested capital, operating and maintenance labor and supervision, maintenance materials, and fuel costs. The present status of technology is such that only an "order of magnitude" estimate can be made for most of these costs, particularly when they are for plants whose proposed construction is many years in the future. Even this type of an estimate requires preparation of a conceptual design of the plants, equipment, processes, and operations in considerable detail. The reactors discussed here have not been studied in the necessary detail. Costs presented for the two-region solution core breeder were obtained by examining a preliminary study of a reactor plant made several years ago and escalating the cost estimates of that study taking into account changes in cost index and experience in reactor equipment fabrication. Power plant costs were based on a comparison with today's cost for conventional plants. Costs for two-region slurry core breeder and one-region slurry converter were obtained by comparison of the plant requirements with those of the reference plant. Although the costs are considerably higher than has been projected by the Homogeneous Reactor Project and others for "nth generation plants" they agree with costs estimated by the PAR Project for a first large commercial plant.

Investments in the three types of plants - excluding fuel inventory, fuel processing and waste facilities and fixed charges based on a 14% annual charge, 0.8 plant factor and 333 MWE were estimated to be the following:

Reactor Type	$M	Plant Cost $/kwE	Fixed Cost Mills/Kwh
Two-region solution core breeder	111,000	334	6.7
Two-region slurry core breeder	107,000	322	6.4
One-region slurry converter	104,000	312	6.2

Details of the cost estimate for the solution core breeder are shown in Table VII-1 and XI-2

Cost of operating the power plants, again excluding the fuel processing and waste facilities, was estimated from personnel requirements to be 0.23 Mills/Kwh for each of the fluid fuel plants, and is assumed here to be independent of type of aqueous reactor. Annual maintenance costs were assessed at 3% of the plant cost for lack of a better number. Total operating and maintenance costs for solution

core, slurry core and one-region reactors are 1.7, 1.6 and 1.5 Mills/Kwh respectively.

On site processing of fuel was specified as being required for all the plants. The minimum cost for a processing plant, including analytical and waste facilities which will be shared by the reactor plant, was judged to be $11,790,000. This plant would be capable of processing and reconstituting 100 kg 1 day or less of thorium and associated uranium from any of the aqueous homogeneous reactors. A cost break-down is shown on page 37 and in Table XI-2. Annual operating and maintenance costs were assumed to be 15% of the plant cost.

For the two-region solution core breeder, higher processing rates are required to obtain short doubling time. An estimate of the relationship of cost to size for on site processing plants is shown below:

Plant capacity, kg/day	<100	300	1,000	4,000
Investment, $M	11,790	14,790	22,000	50,000
Fixed charges, $M/yr	1,650	2,070	3,080	7,000
Operating costs, $M/yr	1,770	2,220	3,300	7,500
Total, $M/yr	3,420	4,290	6,380	14,500
Unit cost, $/kg Th	11,700 kg/d	49	22	12.5

Fuel cycle costs are presented in Table XI-2 as calculated from the fuel cycle data in Table XI-1. They were calculated for several blanket processing rates for the two-region reactors and for 180-day and 60-day residence in processing to show the effect of doubling time on cost. The effect of the short cycles in both reactor and processing is to reduce the inventories at the expense of higher processing costs. It was assumed that the cost in $/kg of processing on the 60-day period would be 1.5 times that estimated for the 180-day period because of the higher radioactivity levels and the necessity for extracting or recycling Pu^{233}. The lowest fuel costs calculated, including fixed charges on the processing, waste and laboratory facilities are 2.5 to 2.7 Mills/Kwh. The 2.7 Mills/Kwh is for the two-region solution core reactor and corresponds to a doubling time of 18 years. Reducing the doubling time to 14, 10 and 6.5 years results in fuel costs of 3.1, 4.1 and 5.6 Mills/Kwh respectively. The minimum doubling time for the two-region slurry core breeder is shown to be 26 years. Reduction of this time requires development of a process for reducing the xenon poison level.

The total energy costs are summarized below:

Reactor	Two-Region Solution Core	Two-Region Slurry Core	One-Region Slurry
Fixed charges on power plant, Mills/Kwh	6.7	6.4	6.2
Operation & Maint of power plant Mills/Kwh	1.7	1.6	1.5
Fixed charges on processing plant, Mills/Kwh	0.89	0.72	0.71
Operation & Maint of processing plant, Mills/Kwh	0.95	0.78	0.76
Inventory charges, Mills/Kwh	1.01	1.07	0.82
Burnup and losses, Mills/Kwh	-0.10	-0.05	0.34
Total energy cost, Mills/Kwh	11.2	10.5	10.3

Increase in power plant size is the factor that should have the largest effect on reducing costs of power from aqueous homogeneous reactor stations. This is because many of the facilities provided for the 333 MWE station would increase very slowly in size with increase in number of reactors and station output at one site. It is premature to attempt to evaluate the effect of station size on cost of the reactor plant; much development is required to establish the feasibility of the systems and whether the costs estimated here are too high or too low. However, the fuel processing is similar enough to existing processes that the effect of plant size on fuel cost can be considered. The total fuel cycle costs including inventory, burnup and losses, and fixed, operating and maintenance costs have been estimated for stations which produce 1000 and 2000 MWE at one site taking into account the effect of processing plant size on the unit cost of processing discussed above. Costs as compared with those for the 333 MWE station are the following:

Station size, MWE	333	1000	2000
Total fuel cycle cost, Mills/Kwh			
Two-region solution core breeder			
10 yr doubling time	4.1	2.5	2.1
14 yr doubling time	3.1	2.0	1.8
18 yr doubling time	2.7	1.7	1.5
Two-region slurry core breeder			
45 yr doubling time	2.5	1.7	1.4
26 yr doubling time	3.5	2.1	1.9
One-region slurry converter	2.6	1.7	1.4

The growth trend in the power industry is toward very large generating stations. in 1950 8% of the electrical generating capacity in the United States was in plants having capacities of 500 to 1000 MW and there were no plants of capacity greater than 1000 MW listed in the Federal Power Commission report. In 1960 the percentages

will be 23 and 11 for 500-1000 MW and 1000 MW plants respectively. Aqueous homogeneous reactors are best suited for the large generating stations where advantage can be taken of large on site processing plant to achieve low fuel cost.

The requirement for processing fuels on site places a cost penalty on the one-region reactor and the two-region slurry core breeder reactors. It has been estimated for the 333 MWE station that the investment in analytical, waste and handling facilities on site could be reduced from the $11,790,000 to $7,000,000 or less. The cost of shipping, processing, and reconstituting the fuel in a central plant having a capacity of 6000 kg/day (investment cost $100,000,000) is estimated to be $25/kg. This reduces the annual charges for processing of the 50 kg/day for the one-region slurry reactor from $3,420,000 to $2,400,000, or less, resulting in a savings of at least 0.4 Mills/Kwh and a reduction in total fuel cost (including fixed charges, operation, and maintenance of on site facilities) from 2.6 to 2.2 Mills/Kwh. Fuel costs for the two-region slurry core breeder would be reduced from 2.5 to 2.2 Mills/Kwh by shipment off site for processing. These savings decrease with increasing size of power plant and are insignificant when the power plant sizes reach 1500 to 2000 MWT.

G. Research and Development Program (Project Director's Appraisal)

There are three important stages in developing a reactor system into a commercial power plant. They are (1) determination of the compatibility of fuel and structural materials and development and demonstration of processes and engineering on a reactor experiment or pilot plant scale; (2) extension of the engineering to large-scale systems and demonstration on a prototype; (3) improvement of equipment and processes and construction of a commercial plant. Step 1 is the important step in determining the technological feasibility; step 2 is the important step in determining the economic feasibility; step 3 is the beginning of exploitation of the system. A schedule for this type of program for the AHR and estimates of the costs are presented in Table XI-3.

Development of the solution fuel system has passed through much of stage 1. Completion of this stage requires that the core tank and solution stability questions be resolved and that HRE-2 be rebuilt with a slurry blanket to demonstrate the feasibility of both core and blanket systems. FY 1961 is the most important year of the program. Solutions to several critical problems must be obtained in that year in order for the program to continue.

1. Analysis of the behavior of the fuel in HRE-2 must be completed, making use of data from HRE-2 and from in-pile loops and autoclaves.

2. The correlation used in predicting corrosion rates for zirconium alloys in-pile must be confirmed by experiments at fluxes approaching those specified for large power reactors.

3. A practical design must be developed for a large reactor vessel. Hydrodynamics experiments for both core and blanket and thorough analysis of the nuclear and mechanical design problems are required.

4. Encouraging progress must be made in developing slurry pumps and valves, and methods for charging and discharging slurry from the blanket of a reactor experiment.

5. Slurry with engineering properties acceptable for a reactor experiment must be demonstrated in test out-of-pile.

6. It must be demonstrated that extended irradiation of the slurry in autoclaves has no serious adverse effects on the properties of the blanket slurry, the activity of the recombination catalyst, or the corrosion of Zircaloy.

7. An acceptable conceptual design must be made for the HRE-2 modifications.

Assuming that satisfactory answers are obtained for the crucial questions in FY 1961, modification of HRE-2 can begin in FY 1962. During the modification period part of the equipment development effort will be aimed at insuring satisfactory operation of equipment and processes for the reactor experiment. Work on fuels and materials will emphasize the investigation of effects of radiation on slurries and container materials in in-pile loops and autoclaves. In-pile work on the solution fuels will be continued but at a reduced level to improve data on the effects of radiation on the solution fuels and on corrosion by those fuels.

The schedule shown requires that a conceptual design be made for the prototype reactor in FY 1962 and that work begin on the development of maintenance tools and methods, circulating pumps, feed pumps, valves, core tank to pressure vessel joints and other items that involve long times for development, procurement or testing. Work must begin on development of special methods and equipment for processing the fuels from the prototype.

Operation of HRE-2 would begin again in FY 1964 and part of the development effort would be associated with the reactor operation. The development effort cannot be projected beyond this time with certainty. It is assumed that design of the prototype would begin early in FY 1965 and construction would begin late in the year. Effort on fuels and materials would be reduced gradually unless unexpected troubles developed. Development effort spent on testing and improvement of equipment, instruments, and maintenance devices would continue at a steady level during the reactor construction. Operation of HRE-2 is shown as being terminated after theee years although it is possible that this operation would be continued to provide additional data for operation of the prototype. The fuel processing section of the prototype plant would begin operation in FY 1968 to prepare fuel for the reactor and the entire plant would be completed in FY 1969. At this time most of the development effort would be associated directly with the plant operation. Design and construction of the first commercial plant could begin any time after FY 1970 with FY 1974 as the earliest completion date.

Programs and schedules for development of one-region slurry converter and two-region slurry core breeder reactors would be similar to that presented above. Problems would be confined to those of the slurry fuel so the costs should be less. The development of a satisfactory recombination catalyst assumes much greater importance in these programs because the power density in the core slurry is much greater than in the blanket slurry. The reactor experiment would begin as a critical experiment to confirm that inhomogeneities in the slurry create no serious nuclear stability problems. Successful operation as a critical experiment would be followed by installation of heat removal equipment to permit operation at power densities approaching those of the large power reactors. The one-region converter reactor would be emphasized in the early development. A two-region slurry core breeder would be a natural outgrowth of the one-region converter.

Table XI-1 Fuel Cycle Data for AHR
(Presented by Project Director)

Reactor Type		Two-Region Solution Core		Two-Region Slurry Core		One-Region Slurry
Total power	MwT	380		1140		1140
Core cycle time	days	400		180	360	480
Blanket cycle time	days	55	137 240	360	720	---
Processing rate	kg Th/day	440	175 100	240	120	50
Reactor systems inventory						
U^{233} + Pa^{233} + U^{235}	kg	74	110 150	800	950	940
Th	met. ton		24	65		24
D_2O	1000 liters		48	135		95
Total inventory-reactor and 180 day processing time						
U^{233} + Pa^{233} + U^{235}	kg	175	210 250	1530	1350	1300
	$M @ $17.50/g	3060	3680 4370	26800	23600	22800
Th	met. ton	100	56 42	106	86	33
	$M @ $22.00/kg	2200	1230 930	2340	1890	730
D_2O	1000 liters		50	135		95
	$M @ $68.00/1		3400	9180		6460
Total	$M	8660	8310 8700	38320	34670	29900
Total inventory - reactor and 60 day processing time						
U^{233} + Pa^{233} + U^{235}	kg	110	145 185	1040	1080	
	$M	1930	2540 3240	18200	18900	
Th	met. ton	50	35 30	79	72	
	$M	1100	770 660	1740	1590	
D_2O	1000 liters		50	135		
	$M		3400	9180		
Total	$M	6430	6710 7300	29120	29670	

Table XI-1 Continued

Reactor Type		Two-Region Solution Core			Two-Region Slurry Core		One-Region Slurry
Gross conversion ratio		1.10	1.09	1.08	108	106	0.92
U^{233} + U^{235} balance							
Gross production	kg/yr	12.2	11.0	9.8	29.3	22.0	-34.6
Processing loss	kg/yr	0.4	0.3	0.2	2.0	1.4	1.1
Net production	kg/yr	11.8	10.7	9.6	27.3	20.6	-35.7
	$M @ $17.50/g	208	187	168	477	360	-625
Th balance							
Burn-up	kg/yr	135	135	135	395	390	340
Processing loss	kg/yr	130	50	30	70	35	20
Net consumption	kg/yr	265	185	165	465	425	360
	$M	5.8	4.1	3.6	10	9.4	7.9
D_2O balance							
Handling loss (2.5%)	1000 l/yr		1.25		3.38		2.38
Processing loss	1000 l/yr	0.12	0.05	0.03	0.06	0.03	0.01
Total loss	1000 l/yr	1.37	1.30	1.28	3.44	3.41	2.39
	$M/yr	93	88	87	234	232	163
System doubling time							
180 day processing time	yr	10	14	18	39	45	
60 day processing time	yr	6.5	10	13	26	36	

Table XI-2 Fuel Costs for AHR
(Presented by Project Director)

Reactor Type		Two-Region Solution Core Breeder							Two-Region Slurry Core Breeder				One-Region Slurry Converter
Processing time	days		180			60			180			60	180
Processing rate	kg Th/d	440	175	100	440	175	100	240	120	240	120		50
Inventory charges													
U^{233} / Pa^{233} / U^{235}	$M/yr @ 4%	122	147	175	77	102	130	1,070	945	728	756		912
	Mills/Kwh	0.16	0.19	0.23	0.10	0.13	0.17	0.46	0.40	0.31	0.32		0.39
Th	$M/yr @ 14%	308	172	130	154	108	93	328	265	244	223		102
	Mills/Kwh	0.40	0.22	0.17	0.20	0.14	0.12	0.14	0.11	0.10	0.10		0.04
D_2O	$M/yr @ 14%	476	476	476	476	476	476	1,285	1,285	1,285	1,285		905
	Mills/Kwh	0.61	0.61	0.61	0.61	0.61	0.61	0.55	0.55	0.55	0.55		0.39
Total	$M/yr	906	795	781	707	686	699	2,683	2,495	2,257	2,264		1,919
	Mills/Kwh	1.17	1.02	1.01	0.91	0.88	0.90	1.14	1.07	0.97	0.97		0.82
Fuel burn-up & losses													
U^{233} / Pa^{233} / U^{235}	$M/yr	-208	-187	-168	-208	-187	-168	-477	-360	-477	-360		625
	Mills/Kwh	-0.27	-0.24	-0.22	-0.27	-0.24	-0.22	-0.20	-0.15	-0.20	-0.15		0.27
Th	$M/yr	6	4	4	6	4	4	10	9	10	9		8
	Mills/Kwh	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00		0.00
D_2O	$M/yr	93	88	87	93	88	87	234	232	234	232		163
	Mills/Kwh	0.12	0.11	0.11	0.12	0.11	0.11	0.10	0.10	0.10	0.10		0.07
Total	$M/yr	-109	- 95	- 77	-109	- 95	- 77	-233	-119	-233	-119		796
	Mills/Kwh	-0.14	-0.12	-0.10	-0.14	-0.12	-0.10	-0.10	-0.05	-0.10	-0.05		-0.34
Fuel processing cost													
Fixed chgs on inv	$M/yr	1,210	840	690	1,830	1,260	1,040	1,960	1,690	2,950	2,530		1,650
	Mills/Kwh	1.56	1.08	0.89	2.36	1.62	1.34	0.84	0.72	1.27	1.08		0.71
Operation & Maint	$M/yr	1,290	900	740	1,960	1,340	1,120	2,100	1,810	3,150	2,720		1,770
	Mills/Kwh	1.66	1.16	0.95	2.52	1.73	1.44	0.90	0.78	1.35	1.17		0.76
Total	$M/yr	2,500	1,740	1,430	3,790	2,600	2,160	4,060	3,500	6,100	5,250		3,420
	Mills/Kwh	3.22	3.24	1.84	4.88	3.35	2.78	1.74	1.50	2.62	2.25		1.47
Total fuel cyc costs	Mills/Kwh	4.2	3.1	2.7	5.6	4.1	3.6	2.8	2.5	3.5	3.2		2.6
Doubling time	years	10	14	18	6.5	10	13	39	45	26	36		--
1,000 MWE	Mills/Kwh	3.0	2.0	1.7	3.8	2.5	2.2	1.9	1.7	2.1	1.9		1.7

Table XI-3 AHR Program
(Presented by Project Director)

	Fiscal Year													
	61	62	63	64	65	66	67	68	69	70	71	72	73	74
Program Objectives														
Resolve feasibility questions of solution core system	——													
Rebuild HRE-2 (Operating funds)	- - -	——	- - -	- - -	- - -	- - -								
Build prototype plant (Capital funds)					- -	——	——	——	——	- - -	- - -			
Build commercial plant (Govt or private funds)											——	——	——	——
Program Costs														
Reactor design and construction, $ million	—3—	—— 3.0 ——			——	——	35 ——							
HRE-2 operation and direct support, $ million			——	——	11 ——									
Prototype operation and direct support, $ million							——	——	——	24 ——	——			
Research and development program, $ million	——	——	——	——	——	——	56 ——	——	——	——	——			
Annual Distribution of Costs														
Fuels and materials research and development - chemistry of fuels, metallurgy, corrosion, effects of radiation on fuels and materials	2.9	3.5	3.0	2.5	2.0	2.0	2.0	1.5	0.9	0.7	0.7			
Systems engineering - equipment, instrument, and systems development, conceptual design and analysis, hydrodynamics studies, engineering research	2.6	3.4	3.5	2.5	3.0	3.0	3.0	3.0	1.0	0.8	0.8			
Fuel processing - development of processes and equipment for manufacturing fuel, separating corrosion and fission products, and handling of wastes. Manufacture of fuel compounds for development program	0.5	1.1	1.0	1.0	1.0	1.0	1.0	0.5	0.2	0.2	0.2			
Plant operations - operation of reactor and processing plants including the design, development and analytical effort required to solve the problems that arise and interpret and report the data obtained from the operation	3.0		1.0	4.0	3.0	3.0	1.0	3.0	7.0	7.0	6.0			
Reactor construction - detailed design, inspection and plant construction		1.0	2.0		1.0	7.0	11.0	12.0	4.0					

XII. ACKNOWLEDGEMENTS

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APPENDIX A

Excerpt of section of "Ad Hoc Advisory Committee's Report on Reactor Policies and Programs" on "Appraisal of Present Status and Program" of "Fluid Fuel Reactors."

"Fluid Fuel Reactors

"There are two principal reasons for interest in fluid fuel reactors. One of these is to supplement the preponderance of work on heterogeneous or solid fuel reactors to give added insurance for achieving low cost nuclear power through high temperature or simplicity of the fuel cycle, or both. The second reason is that fluid fuel systems appear to offer the only possibility for thermal breeding on the thorium U-233 fuel cycle with an acceptable doubling time -- say ten years or less. Uncertainties in the value of eta (the number of neutrons emitted per neutron absorbed) make it impossible to give positive assurance that breeding on this cycle can, in fact, be achieved. The Committee strongly recommends that urgent effort be devoted to measurements of eta for U-233 by various laboratories and by various methods.

"Until better knowledge of eta is obtained, work on thermal breeders should be held to a minimum. However, development of a thermal breeder reactor, if shown to be feasible, should be included as part of the overall program at approximately the present level of effort on the aqueous homogeneous project. Based on present information, breeding with the thorium U-233 cycle is uncertain for aqueous homogeneous reactors, even more uncertain for liquid metal fuel reactors, and not possible for molten salt reactors.

"Of the three fluid fuel reactor types, the aqueous homogeneous is of interest as a breeder and because of its simplified fuel cycle, the molten salt is of interest because of high temperature operation and simplified fuel cycle, and the liquid bismuth because it might incorporate all three advantages. All three types, particularly the liquid bismuth, present many difficult development problems. For example, it is probable that for successful breeding in the aqueous homogeneous or in the liquid bismuth reactors, a slurry will have to be used at least in a blanket, and the development of adequate slurry technology is a very difficult job.

"The three fluid fuel systems should be evaluated first as to promise for producing low cost nuclear power and not more than one project supported on the basis of that objective. When additional information on the value of eta for U-233 is available, the situation with respect to thermal breeding should be re-examined, with a view toward increased support. We see no present basis for the construction of fluid fuel reactors of any of the three types."

APPENDIX B

List of Principal Items Considered To Be Included in Power Plant Investment For Power Station With Fluid Fuel Reactors Shown in Table VII-2

310 LAND & LAND RIGHTS

311 STRUCTURES & IMPROVEMENTS

.1 Site Improvements
.11 Site Preparation
.12 Site Drainage
.13 Final Grading & Landscaping

.2 Site Facilities (General Use Facilities)
.21 Access Railroad & Yard Trackage
.22 Access Road, Plant Roads, Parking Areas & Walks
.23 Fences & Gates
.24 Yard Lighting System
.25 Intrasite Communication System
.26 Service Water System
.261 Wells & Pumps
.262 Elevated Storage Tank
.263 Service Water Piping
.264 Fire Protection System
.265 Domestic Water Treatment & Distribution
.27 Sanitary Sewer System

.3 Station Buildings
.31 Nuclear-Steam Generator Building
.311 Excavation & Backfill
.312 Concrete work (incl. Shielding & Floors)
.313 Structural & Misc. Steel
.314 Encls. (Walls & Ceil. other than concrete)
.315 Floor Covering
.316 Doors & Windows
.317 Roofing
.318 Painting
.319 Building Services

.3191	Plumbing & Drain. Facilities
.3192	Lighting Facilities
.3193	Heating & Air Conditioning
.3194	Ventilation Facilities
.3195	Elevators
.3196	Building Crane
.3197	Hoists
.3198	Fire Protection Equipment
.32	Service & Maintenance Building
.321	Excavation & Backfill
.322	Concrete Work (incl. Floors)
.323	Structural & Misc. Steel
.324	Encls. (Walls & Ceil. other than concrete)
.325	Floor Covering
.326	Doors & Windows
.327	Roofing
.328	Painting
.329	Building Services
.3291	Plumbing & Drain. Facil.
.3292	Lighting Facil.
.3293	Heating, Ventil., & Air Cond.
.3294	Building Crane
.3295	Kitchen Equipment
.3296	Fire Protection
.33	Turbine & Auxiliaries Building
.331	Excavation & Backfill
.332	Found. & Concrete (incl. Floors)
.333	Structural & Misc. Steel
.334	Encls. (Walls & Ceil. other than concrete)
.335	Doors & Windows
.336	Painting
.337	Building Services
.3371	Plumbing & Drainage Facilities
.3372	Lighting Facilities
.3373	Heating, Vent., & Air Cond.
.3374	Personnel Lift
.3375	Hoists
.3376	Fire Protection Equip.
.34	Chemical & Waste Buildings (Listed Separately)
.35	Other Station Buildings
.351	Warehouse
.352	Water Treatment Building
.353	Chlorine Cylinder Storage Shed

.354 Gas Cylinder Storage Shed
.355 Combustibles Storage Bldg.
.356 Gate House

312 REACTOR & STEAM GENERATOR PLANT

.1 Nuclear Reactor Equipment
.11 Core Vessel
.12 Core Graphite
.13 Thermal Shield
.14 Reactor Vessel (Pressure Shell)
.15 Control Rods & Misc. Hardware
.16 Supports & Shield

.2 Heat Transfer System
.21 Primary (Fuel-Coolant) System
.21a Fill & Storage Equipment
.21a1 Melt Tank
.21a2 Filter
.21a3 Flow Control Valve
.21a4 Freeze Valve
.21a5 Storage (drain-dump) Tanks
.21a6 Storage Tank Internals
.21a7 Fill Pumps
.21a8 Pressure Syphon Fill System (incl. Valves)
.21a9 Dump Valves
.21a10 Fuel Addition-Enrichment Equipment

.21b Primary Loop Equipment
.21b1 Primary Loop Pumps
.21b2 Blanket Loop Pumps
.21b3 Primary Loop Heat Exchangers
.21b4 Blanket Loop Heat Exchangers
.21b5 Primary Loop Pressurizers
.21b6 Blanket Loop Pressurizers
.21b7 Primary Loop Check Valves
.21b8 Blanket Loop Check Valves
.21b9 Fuel Sampler
.21b10 Remainder of System

.22 Intermediate (Coolant) System
.22a Fill & Storage Equipment
.22a1 Filter
.22a2 Fill Valve
.22a3 Fill Pump
.22a4 Storage Tanks
.22a5 Dump Valve

.22b Intermediate Loop Equipment
.22b1 Intermediate Loop Pumps
.22b2 Expansion Tanks
.22b3 Superheaters
.22b4 Reheaters
.22b5 Cold Trap Precipitators
.22b6 Cold Trap Heat Exchangers
.22b7 Plugging Indicators

.3 Reactor Plant Auxiliaries
.31 Radioactivity Containment-Confinement
.311 Reactor Vessel Containment
.312 Off-Gas System Containment
.313 Fuel Fluid (drain-dump) Tanks Containment
.314 Fuel Fluid Fill Pumps Containment
.315 Primary Loop Pumps Containment
.316 Primary Loop Pressurizers Containment
.317 Fuel Fluid Valves Containment
.318 Fuel Fluid Piping Containment
.319 Fuel Addition-Enrichment Equip. Containment

.32 Let-Down & Recombiner Equipment
.321a Core Fuel Gas Separators
.321b Blanket Fuel Gas Separators
.322a Core Fuel Let-Down Economizers
.322b Blanket Fuel Let-Down Economizers
.323a Core Fuel Led-Down Coolers
.323b Blanket Fuel Let-Down Coolers
.324a Core Fuel Evaporators
.324b Blanket Fuel Evaporators
.325a Core Fuel Entrainment Separators
.325b Blanket Fuel Entrainment Separators
.326a Core Fuel Recombiners
.326b Blanket Fuel Recombiners
.327a Core Fuel Condensers
.327b Blanket Fuel Condensers
.328 Condensate Storage Tanks

.33 Reactor Off-Gas Equipment
.331 Hold-Up Tanks
.332 Cold Traps
.333 FP Absorbers
.334 Vacuum Pumps
.335 Compressors
.336 Hg Seal Pots
.337 Filters-Purification Equipment
.338 Heat Exchangers

.34	Inert Gas Equipment
.35	Purge Equipment
.351	Purge Coolers
.352	Purge Pumps
.353	Purge Pump Coolers
.36	Heating, Cooling, & Ventilating Equipment
.361	Primary System Heating & Cooling Equipment
.361a	Reactor Heating & Cooling Equipment
.361a1	Helium Blowers
.361a2	Electric Heaters
.361a3	Heat Exchangers (Water or Dowtherm Cool.)
.361b	Fill & Storage Heating & Cooling Equip.
.361b1	Helium Blowers
.361b2	Electric Heaters
.361b3	Heat Exchangers (Water or Dowtherm Cool.)
.361c	Primary Loop Pumps & Pipe Heat. & Cool. Equip.
.361c1	Helium Blowers
.361c2	Electric Heaters
.361c3	Heat Exchangers (Water or Dowtherm Cool.)
.361d	Primary Heat Exchangers Heat. & Cool. Equip.
.361d1	Helium Blowers
.361d2	Electric Heaters
.361d3	Heat Exchangers (Water or Dowtherm Cool.)
.362	Intermediate System Startup Heating Equipment
.363	Cell Cooling Equipment (closed system)
.3631	"Fin-Fan" Cooling Units
.3632	Water Circulating Pumps
.3633	Heat Exchanger (water coolers)
.3634	Storage (dump) Tank
.364	Shield Cooling Equipment (closed system)
.3641	Thermo Panel Coils
.3642	Water Circulating Pumps
.3643	Heat Exchanger
.3644	Storage (dump) Tank
.365	Raw Water Cooling System
.3651	Pumps
.366	Ventilation System
.3661	Fans
.3662	Filters
.3663	Ductwork

.4 Steam Generator Equipment
.41 Steam Generator (Boiler)
.42 Loeffler System Components
.421 Steam Pumps
.422 Evaporator Drums

.5 Steam Generator Auxiliaries
.51 Boiler Feedwater Equipment
.511 Feedwater Heaters
.512 Boiler Feed Pumps
.513 Deaerator & Aux. (incl. Pumps)
.514 Condensate Transfer Pumps
.515 Heater Drain Pumps
.516 Condensate Storage Tanks
.517 Reducing & Desuperheating Station

.52 Feedwater Testing Equipment
.521 Demineralizing Equipment
.522 Chemical Pumps & Tanks
.523 Laboratory Equipment

.53 Service Boiler & Fuel Oil Equipment
.531 Service (oil burning) Boiler
.532 Service Oil Unloading & Transfer Pumps
.533 Service Oil Storage Tank

.6 Reactor & Steam Generator Piping Systems (incl. Valves)
.6a Primary Fuel-Coolant Fill & Storage Piping
.6b Primary Heat Transfer Loop Piping
.6c Intermediate Coolant Fill & Storage Piping
.6d Intermediate Coolant Loop Piping
.6e Loeffler System Piping
.6f Off-Gas System Piping
.6g Inert Gas System Piping
.6h Reactor Heating & Cooling Piping
.6i Fill & Storage Heating & Cooling Piping
.6j Primary Pump & Pipe Heating & Cooling Piping
.6k Primary Heat Exchanger Heating & Cooling Piping
.6l Cell Cooling System Piping
.6m Shield Cooling System Piping
.6n Raw Water Cooling System Piping
.6o Steam System Piping
.6p Feedwater System Piping
.6q Feedwater Treating System Piping
.6r Service Oil System Piping

.7	Reactor & Steam Generator Plant Insulation
.7a	Reactor Vessel Contain. & Heat. & Cool. Equip.
.7b	Off-Gas System Contain. & Heat. & Cool. Equip.
.7c	Fuel-Fluid Fill & Stg. Con. & Heat. & Cool. Equip.
.7d	Prim. Loop Pump & Pipe Contain. & Heat. & Cool. Equip.
.7e	Prim. Heat Exch. & Heat. & Cool. Equip.
.7f	Interm. System Equip. & Piping & Heat-Up Equip.
.7g	Cell Cooling Equipment & Piping
.7h	Shield Cooling Equipment & Piping
.7i	Steam Generator Equipment
.7j	Steam System Piping
.7k	Feedwater System Equip. & Piping
.8	Controls & Instrumentation Equipment
.81	Panels & Consoles
.82	Central Control System
.83	Primary Loop System
.84	Intermediate Loop System
.85	Steam Loop (incl. Loeffler) System
.86	Auxiliary Systems
.86a	Nuclear Instrument & Health Monitors
.86b	Inert & Off-Gas System
.86c	Fuel Fluid Fill & Stg. System
.86d	Reactor Heating & Cooling System
.86e	Fuel Fluid Fill & Stg. Heat. & Cool. System
.86f	Primary Loop Pump & Pipe Heat. & Cool. System
.86g	Primary Heat Exch. Heat. & Cool. System
.86h	Intermed. Loop Startup Heat System
.86i	Cell Cooling System
.86j	Shield Cooling System
.86k	Raw Water System
.86l	Ventilation System
.86m	Feedwater System
.86n	Feedwater Treating System
.86o	Elec. & Penumatic Equipment
.9	Hot Cell & Remote Maintenance Equipment
.91	Hot Cell Equipment
.92	Remote Maintenance Equipment

314 TURBINE-GENERATOR PLANT EQUIPMENT, ETC.

.1	Turbine Foundation
.2	Turbine-Generator & Exciter

.3 Turbine-Plant Auxiliaries
.31 Lubricating Oil Equipment
.32 Hydrogen Cooling Equipment
.33 Gland Seal Equipment
.34 Closed System Cooling Water Equipment
.35 Turbine Room Instruments & Control Panel
.36 Crane & Rails
.37 Turbo-Generator Shelter

.4 Turbine Plant Piping
.41 General Piping
.42 Insulation

.5 Condenser & Auxiliaries
.51 Condenser
.52 Circulating Water Pumps
.53 Condensate Pumps
.54 Vacuum Pumps

.6 Circulating Water System
.61 River Intake Rack Structure
.611 Piling & Fender Structures
.612 Structural Steel Rack
.62 Screen & Pump Chamber
.621 Excavation & Backfill
.622 Concrete Structure
.623 Equipment
.6231 Traveling Screens
.6232 Stoplogs
.6233 Pumps & Piping
.6234 Chlorination Equipment
.63 Intake & Discharge Pipe
.631 Reinforced Concrete Pipe
.632 Steel Pipe, Valves & Connections
.64 Discharge Seal Well Structure
.65 Discharge Canal
.651 Excavation
.652 Blanket & Riprap
.66 Discharge Structure
.67 River Bank Revetment

315 ACCESSORY ELECTRICAL EQUIPMENT

.1 Foundations & Structures
.11 Generator Leads Foundations & Supports
.12 Equipment Foundations
.13 Manholes & Handholes

.2 Power & Conversion Equipment
.21 Station Transformers
.211 Station Service Transformers
.212 Lighting Transformers
.22 Emergency Power Facilities
.221 Motor Generating Equipment
.23 D-C Power Equipment
.231 Storage Batteries
.232 Battery Charging Equipment

.3 Conduits, Conductors, & Insulators
.31 Conduits, Ducts, & Trays
.311 Underground Conduits & Ducts
.312 Exposed Conduits & Trays
.32 Conductors & Insulators
.321 Generator Leads
.322 Starter Transf. Feeders
.333 Power Wiring
.334 Control Wiring
.335 Miscellaneous

.4 Switching, Control, & Protective Equipment
.41 Control Boards & Panels
.411 Main Control Board for BTG
.412 Load Frequency Control Equipment
.413 A-C & D-C Distribution Panels
.42 Metal-Clad Switchgear
.421 4160 V. Station Auxiliary
.422 480 V. Station Service
.43 Other Control Equipment
.431 Motor Control Load Centers
.432 Misc. Remote Control Equipment
.433 Fault Protectors
.434 Indicators

.5 Station Grounding System

316 MISCELLANEOUS POWER PLANT EQUIPMENT

.1 Compressed Air Equipment
.2 General Shop & Maint. Equipment
.3 Office Furniture & Fixtures
.4 Stores Fixtures and Equipment
.5 First Aid Equipment
.6 Fire Extinguishing Equipment
.7 Miscellaneous

SUGGESTED LIST
ELEMENTS OF DISTRIBUTIVE COST

DIRECT CONSTRUCTION COST (Direct Material & Labor)

INDIRECT CONSTRUCTION COSTS
- Temporary Construction Buildings, Utilities, & Facilities
- Construction Equipment & Small Tools
- Office Equipment, Supplies, & Expense
- Warehousing Expense
- Local Sales & Use Taxes
- Payroll Insurance & Taxes (Workman's Comp., Soc.Sec, Tax, etc.)
- Incidental Labor Expense (recommitment, bargaining, weld. qualif., etc.)
- Technical Services (inspection, laboratory tests, etc.)
- Field Supervisory & Clerical Payroll (all indirect on site labor)
- Construction Permits & Licenses
- Performance Bond

OVERHEAD CONSTRUCTION COSTS
- Preliminary Studies & Investigations
- Engineering & Design
- Legal Fees & Taxes During Construction
- Insurances During Construction (other than payroll insurance)
- Injuries & Damages Not Covered by Insurance (floods, land damages, etc.)
- Contractors' Fees (home office expense and profit)
- Start-Up & Preliminary Operation Expense (prior to commercial operation)
- Administrative & General Expense
- Interest During Construction

ALLOWANCE FOR OMISSIONS & CONTINGENCIES (on everything above)

TOTAL CONSTRUCTION COST (sum of above)