MASTER

STUDY OF REMOTE MILITARY POWER APPLICATIONS
REPORT NO. 12
EVALUATION AND SELECTION OF APPLICABLE REACTOR CONCEPTS
FOR
UNITED STATES ATOMIC ENERGY COMMISSION
NEW YORK OPERATIONS OFFICE
NEW YORK, NEW YORK

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Contract No. AT (30-1)-2441

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SECTION I

INTRODUCTION

On August 17, 1959, the United States Atomic Energy Commission, through the New York Operations Office, issued Contract No. AT (30-1)-2441 to Kaiser Engineers for the "Study of Remote Military Applications". The study is essentially an economic evaluation of the construction and operation of nuclear power plants at several designated military installations, where increased power generating capabilities of 5 to 40 mwe may be required by the Government for the period 1963 through 1970.

The purpose of this report is twofold: First, evaluate and select reactor concepts suitable for power generation within the 5,000 to 40,000 kw range, projected for the remote military installations being studied, and which can be in operation in the mid 1960's; second, for those reactor concepts selected for final consideration, outline the economic and technical characteristics to permit the selection of a specific reactor for each application.

In selecting nuclear power plants for the specific applications, the following evaluation criteria have been established:

1. Reactor selection and design will be based on technical feasibility, economics, safety, and reliability. Relative weights given to these factors will be evaluated on a site-by-site basis.
2. Reactor types to be considered will be limited to those systems which are technically well-defined and proved at present, or which can be reasonably expected to be proved by the time construction would begin.
3. No fossil fuel fired superheater will be used.
4. Choice of enrichment will be based on economic considerations only.
5. Annual fixed charges will be approximately 5%. This is based on straight line depreciation for the life of the plant but not to exceed 20 years. No interest is to be included (Ref. 31).
6. Fuel cycle costs will be based on long range civilian reactor economics except that no fuel use charge or interest on core fabrication will be charged. Plutonium will be credited at \$12/gm (Ref. 31).

Total power costs and detail economics for specific applications are contained in the individual site study reports, and not in this evaluation and selection report.

This report is No. 12 of a series of reports completed under Contract No. AT (30-1)-2441. The completed study comprises a Summary Report and a number of site reports in addition to this report.

SECTION II

SUMMARY AND CONCLUSIONS

To select those reactors which best meet the criteria for the nuclear power plants for remote military bases, a brief evaluation was made of the major reactor concepts presently under development. These concepts include the water cooled and moderated reactors - both direct and indirect cycle, organic cooled and moderated reactors, heavy water cooled and moderated reactors, gas cooled reactors, sodium cooled graphite moderated reactors, fast reactors, and fluid fuel reactors. Each evaluation consisted of a study of the basic reactor characteristics, inherent advantages and disadvantages; technological status; plant economics for construction in the early 1960's; and the future potential for the particular reactor concept. These reviews are summarized in Section III.

In examining the eight concepts listed above, it became apparent that three concepts, sodium cooled graphite moderated, fast breeder reactors, and fluid fuel reactors, were less suitable than the others for power plants at remote military installations. The remaining five were evaluated further. The evaluations of these five reactor concepts are set forth in Appendices A through E.

As a result of these reviews, it was concluded that two reactor concepts are suitable for those applications requiring power before 1965. The economics of these concepts, direct cycle boiling water reactors and pressurized water reactors, are discussed in Section IV. Estimated capital and fuel costs are summarized below:

<u>Power Level</u>	<u>Pressurized Water Reactor</u>		<u>Boiling Water Reactor</u>	
	<u>Capital Cost</u> <u>\$/net kw</u>	<u>Fuel Cost</u> <u>mills/kwhr</u>	<u>Capital Cost</u> <u>\$/net kw</u>	<u>Fuel Cost</u> <u>mills/kwhr</u>
5,000 kwe	750-1100	7.5-8.5	750-1000	7.5-8.5
10,000 kwe	700-1000	5.5-8.0	650- 850	5.5-8.0
20,000 kwe	500- 750	4.5-6.5	500- 650	4.5-6.5
40,000 kwe	400- 550	3.0-4.5	350- 500	3.0-4.5

It was concluded that organic cooled and moderated reactors would be suitable only for plants to be in operation in 1965 and later. Since organic reactors using presently developed fuel elements (U alloys clad with aluminum) offer no economic advantage over boiling and pressurized water reactors, their less demonstrated technology makes them relatively unattractive for use at remote

sites. However, by 1965, UO_2 fuel elements suitable for organic cooled reactors probably will be developed and the background of experience in this type of reactor will be substantially increased.

Heavy water cooled and moderated reactors, gas cooled reactors, sodium graphite reactors, and fast breeder reactors are not considered suitable because of their high capital costs and limited experience compared with light water reactors. Fluid fuel reactors are also not considered suitable because they are not technically proved.

Section V discusses the different criteria which will be applied to land based nuclear power plants integrated with existing power systems, isolated land based plants, barge mounted mobile power plants and "one trip" barge mounted power plants.

SECTION III

REACTOR EVALUATION AND SELECTION

The following summarizes the review of the reactor concepts and indicates those concepts selected for further consideration.

A. Water Cooled and Moderated Reactors - Direct Cycle

Only boiling light water reactors are included in this class of reactor. The principal advantages of this reactor are its simplicity and high pressure steam production compared with pressurized water reactors. The main disadvantage is the use of radioactive steam in the turbine.

Three boiling water prototype reactors are in operational status, several boiling reactor experiments have been operated, one large-scale boiling water power plant (Dresden) has recently commenced operation and should be operating at design power level by the end of 1960, several other reactors of this type are in various stages of design, development and construction. Experience to date with boiling water direct cycle systems has demonstrated that radioactive steam can be used in the turbine with no difficulty and the reactor, moreover, is inherently self-regulating. Economically, direct cycle boiling reactors are at least as attractive as any other reactor concept at the present time. A description of the economics of this reactor concept is contained in Section IV. The potential for further reduction in the cost of power from the bwr is better than for most other presently established reactor concepts.

This reactor concept has been selected as being suitable for the power plant applications of this study.

For more detail on boiling water reactors, see Appendix A.

B. Water Cooled and Moderated Reactors - Indirect Cycle

Both the pressurized water reactors and the closed cycle boiling reactors are included in this category. Although two closed cycle boiling reactor plants are presently being constructed, there is little evidence that these plants have any advantage over direct cycle boiling water reactors or pressurized water indirect cycle reactors.

Special advantages of the pressurized water reactor are its compactness and inherent load following characteristics.

The experience with pressurized water reactors is more extensive than with any other reactor concept. Ten to twelve pressurized water reactors for the propulsion of naval vessels are presently in operation. In the power generation field, both the Shippingport Plant and the Army Package Power Reactor (SM-1) have been operating satisfactorily for the past few years. Although no pressurized water reactors for power production have been built in the United States within the size range of interest to this study, the recent "Task Force Evaluation Report for the Small-Sized Nuclear Power Plant Program" (Ref. 1) estimated that pressurized water reactors would be only slightly less economical than boiling water reactors in the 20 mwe power range. A description of the economics of this reactor concept is contained in Section IV. The potential for future reductions in capital and fuel costs in pressurized water reactors is not too different from that of boiling reactors.

This reactor concept has been selected as being suitable for the power plant applications of this study.

For a further discussion of this concept, see Appendix B.

C. Organic Cooled and Moderated Reactors

This type of reactor is similar to the pressurized water reactor except that an organic material (usually a polyphenyl compound) is substituted for water as the coolant and moderator. The advantages of this concept are low vapor pressure of the organic material at elevated temperatures (e.g., an organic system pressure of 120 psig is adequate for producing 800 psig steam), favorable corrosion characteristics, and the feasibility of generating superheated steam. Principal disadvantages are thermal and radiolytic decomposition and poor heat transfer properties of the organic material.

The only operating experience to date with an organic cooled and moderated reactor (ocmr) is the 16 mwt Organic Moderated Reactor Experiment (OMRE). A 10-15 mwe nuclear power plant prototype is presently under construction at Piqua, Ohio and a larger experimental reactor is also planned.

Predicted unit power costs of organic nuclear power plants in the 5,000 to 40,000 kwe size range are not too different from the boiling and pressurized water reactors at the present time. However, fuel cycle costs are subject to a fairly high degree of uncertainty since the present ocmr concept is dependent on the use of metallic uranium alloy fuel at temperatures in which "swelling" of the uranium due to radiation damage is a significant problem. Should present programs aimed at developing improved fuel materials (UO_2 in a cladding of "Aluminum

Powdered Metal") for the organic moderated reactor prove successful, the economics of this concept would be greatly improved.

For the power plants at remote sites under consideration in this study which must be in operation in 1963 or 1964, organic moderated reactors will not be suitable. Under the ground rules of this study, the low capital cost for organic reactors is more than offset by the higher fuel and organic makeup costs using metallic uranium fuel elements. Consequently, total power costs are not estimated to be any lower than the boiling and pressurized water reactors and there is no incentive for using the less demonstrated ocmr. However, the organic moderated reactor should be considered further for plants which are to be in operation in 1965 or later. This will permit approximately two years of additional research and development. It will also permit the operating experience of the City of Piqua reactor, which will begin operation in 1961, to be factored into design.

For a more detailed discussion of this concept, see Appendix C.

D. Heavy Water Cooled and Moderated Reactors

Included in this category are reactors using heavy water as a moderator with either light or heavy water cooling. Heavy water moderated gas cooled reactors are discussed under "Gas Cooled Reactors" on page 8.

The principal advantage of heavy water reactors is the low neutron absorption of the moderator which, in large sizes, permits the use of natural uranium fuel.

No heavy water moderated power reactors have been operated in the United States. However, a boiling heavy water process heat reactor is presently in operation in Norway, and several heavy water reactors are in operation for research and Special Nuclear Material production in the United States, Canada and elsewhere. Several heavy water power reactors are presently planned or under construction in both the United States and Canada.

The capital cost of heavy water reactors in the 5 to 40 mwe size range is substantially higher than light water reactors. Moreover, in this size range it does not appear that fuel costs of a heavy water reactor will be any lower than light water reactors, while operating and maintenance costs should be somewhat higher.

Because of the higher capital cost for this range of power and the lack of operating experience, this concept is not considered suitable for the power plant applications of this study.

For further reference to this concept see Appendix D.

E. Gas Cooled Reactors - Direct Cycle and Indirect Cycle

Direct cycle gas cooled reactors generate power directly from the gas coolant by means of a gas turbine prime mover. Indirect cycle plants transfer the heat to water, generating steam for use in a conventional steam turbine.

The advantage of gas cooled reactors is the high temperatures and consequently excellent steam conditions achievable.

A disadvantage is the relatively poor heat removal capability which results in low power density and relatively large physical size.

No gas cooled reactors have been operated in the United States (to generate power), although several are in operation abroad. The Gas Cooled Reactor Experiment (GCRE), a small gas cooled reactor intended for ultimate use in a direct cycle power plant, is presently in operation. A special gas cooled reactor application is the direct cycle Aircraft Nuclear Propulsion Reactor. A land-based prototype of this reactor has been operated. Three indirect cycle plants are presently planned or under construction; two use graphite as the moderator and one uses heavy water.

The indirect cycle gas cooled reactors presently being developed show economic potential only in large plant capacities. Capital cost in the 5 to 40 mwe size range is considerably higher than the light water moderated reactors with no apparent saving in fuel or operating and maintenance cost to offset this difference.

A more detailed discussion of this concept is contained in Appendix E.

Because of the higher capital cost and the lack of experience, this concept is not considered suitable for the power plant applications of this study.

F. Sodium Cooled, Graphite Moderated Reactors

This reactor concept makes use of the excellent heat transfer capability of sodium in a nuclear reactor in which most of the fissions are caused by thermal neutrons. The principal advantage compared to water reactors is the excellent steam conditions which are achievable, comparable to those used in

modern fossil fueled steam plants. A disadvantage is the complexity of the plant which results from the chemical reactivity of sodium with respect to both air and water as well as long-lived induced activity from Na^{24} .

The only sodium graphite reactor in operation in this country is the 6 mwe SRE. A 75 mwe prototype sodium graphite reactor power plant is presently under construction. While the economics of sodium graphite reactors of advanced design may be attractive in large capacity plants of more than 100 mwe, the complexity and large physical size of the system probably will always lead to higher capital costs in the 5 to 40 mwe range for sodium graphite reactors than for light water reactors of the same size. At the present time, fuel, operating and maintenance costs of the sodium graphite reactors are estimated to be higher than comparable light water reactors. However, development of uranium carbide fuel material might improve significantly the fuel cost of sodium graphite reactors.

Because of the higher capital cost and the lack of experience, this concept is not considered suitable for the power plant applications of this study.

G. Fast Breeder Reactors

This class of reactors includes those in which most of the fission is caused by high energy, or fast, neutrons. In a fast reactor, it is possible to "breed" more fissionable material than is consumed - a characteristic which constitutes the principal advantage of this type of reactor. Also, since most fast reactors use liquid metal coolants (usually sodium), it is possible to achieve high temperatures and correspondingly high thermal efficiencies.

One disadvantage of the fast reactor concept is the extremely high fissionable material inventory required to sustain criticality. Another disadvantage is the high fuel fabrication and reprocessing costs which result from the short burn-ups achievable. In addition, there are the problems in using sodium as the coolant as indicated in the previous discussion on sodium cooled reactors.

The principal operating experience on fast reactors has been obtained from the operation of the EBR-1. Two other fast reactor prototypes are presently under construction in the United States; one of 16 mwe capacity and one of 100 mwe capacity.

Capital cost of both of these plants is considerably higher than light water reactors of a comparable capacity. Although the fuel burn-up cost theoretically can be reduced to zero in a fast reactor, the high fuel fabricating and processing costs result in net fuel costs at least as high as the light water reactors. The development of pyro-metallurgical reprocessing schemes may reduce fuel costs substantially for fast reactors. It is likely, however, that the complexity of sodium cooled fast reactors will always lead to high capital costs in the 5 to 40 mwe size range.

Because of the higher capital cost and the limited experience, this concept is not considered suitable for the power plant applications of this study.

H. Fluid Fueled Reactors

This category includes three concepts: aqueous homogeneous reactors, liquid metal fueled reactors and fused salt fueled reactors. All three concepts involve fluid fuel materials which can be circulated.

The advantages of all three reactor concepts are the elimination of fuel element fabrication and less costly fuel processing.

All three have the same disadvantage - substantial corrosion-erosion effects and material handling problems with the fluid fuel mixture.

Four aqueous homogeneous reactor experiments and a fused salt reactor experiment have been operated for short periods. The undeveloped state of the three types of fluid reactors makes a discussion of their economics highly speculative.

Since fluid fueled reactors are not technically proved, nor can they reasonably be expected to be proved by the time construction would begin for the applications being studied, these reactors will be given no further consideration.

SECTION IV

ECONOMICS OF THE SELECTED REACTOR CONCEPTS

A. Capital Cost

The capital costs of direct cycle boiling water reactor power plants as a function of power level, in the range of interest (5 to 50 mwe), are shown in Figure 1, page 12. Because of uncertainties in the cost estimates and variations between manufacturers, a range of costs for a given capacity is indicated rather than a single value. The dashed line represents an average value for boiling water reactors and is consistent with a series of estimates by a leading boiling water reactor manufacturer for plants within this range (Ref. 29). The latter is based upon experience with two plants already completed, as well as several which are presently in an advanced stage of design.

Figure 2, page 13, shows the capital costs of pressurized water reactor power plants as a function of power level. The range in cost estimates for a given power level is greater than for boiling water reactors. The upper limit of the band represents estimates such as those prepared by a leading pwr manufacturer (Ref. 2). The lower limit represents estimates such as those prepared by another prominent pwr manufacturer (Ref. 3). The dashed line shown in Figure 2 represents an average cost of pwr reactors in this range. The large difference between the two sets of cost estimates for pressurized water reactor power plants is difficult to explain. Part of the difference can be accounted for by the fact that one plant achieves lower fuel cost by the use of relatively large (approximately 1/2" diameter) fuel rods at the expense of somewhat lower power density and consequently higher capital cost (see Section IV B). Another cause for this difference is that the higher cost estimates are based on a more conservative design philosophy than the other estimates. To cite only a few examples where this difference in design philosophy is apparent, one design provides only a few storage tanks as a "waste disposal system" whereas the other provides for an elaborate waste treatment and concentration system; one design uses a single primary pump, the comparable more costly design uses two; one design provides for a "safety injection system" to remove reactor heat after a complete loss of cooling, the other provides no "last-ditch" cooling system.

There is not yet sufficient experience with pwr power plants to determine whether the additional cost associated with the more conservative design will be justified by a corresponding

FIGURE 1
CAPITAL COSTS
BOILING WATER REACTOR POWER PLANT
(BASED ON 1959 COSTS)

FIGURE 2

CAPITAL COSTS

PRESSURIZED WATER REACTOR POWER PLANT

(BASED ON 1959 COSTS)

reduction in operating and maintenance costs or increased plant availability and reliability.

The U. S. AEC and the Dept. of Defense are jointly financing the development and design of an approximately 7,000 kwe pressurized water reactor power plant which has been designated the SM-2. This plant is a scaled-up and improved version of the SM-1 (formerly APPR-1) plant presently operating at Ft. Belvoir, Virginia. The responsibility for the development and the design effort, estimated at approximately 3.5 million dollars has been assigned to Alco Products, Inc. This effort will culminate in detailed plans and specifications for the entire plant. An estimated construction cost for this plant at a U. S. site is approximately \$4,000,000 excluding design and development costs. This construction cost is substantially less than any costs received from other nuclear manufacturers for reactors in the 5 - 7,000 kwe range (see Figure 2). One of the primary reasons for the higher cost of plants using commercial designs in the 5 - 7 mw range, compared with the SM-2, is that the former contain substantial sums for design, development and testing. For the SM-2, these costs are being written off in the present Alco program.

B. Fuel Costs

Figure 3, page 16, is a plot of fuel costs for both boiling and pressurized water reactors as a function of net electrical output. It can be seen that fuel cycle costs decrease as plant capacity increases as a result of the better neutron economy achievable in a large reactor.

There is nothing inherent in the boiling and pressurized water reactor concepts that would result in substantial differences in their fuel cycle costs. All important aspects of fuel cycle costs are approximately the same for both reactors:

1. Both reactors can use metal-clad UO_2 fuel elements of about the same diameter; achievable burnup should be approximately the same.
2. Both reactors use Zircaloy, stainless steel or aluminum alloy as the fuel element cladding.
3. Both reactors use approximately the same enrichment for a given cladding material and have approximately the same conversion ratio.

As can be seen in Figure 3, there is little difference in the fuel costs of certain pwr and bwr designs.

However, in comparing other specific designs, there can be substantial differences in fuel cycle costs. Changes in fuel element geometry and cladding can cause significant variations in over-all fuel costs. For example, the cost per kilogram of contained uranium for fabricating fuel consisting of large diameter fuel pins is significantly less than the cost of fabricating fuel with smaller diameter pins. On the other hand, reducing fuel pin diameter results in higher achievable power density, especially in pressurized water reactors, and reduces the size of the reactor pressure vessel and shielding. Thus, in any given design, it is necessary to choose an "optimum" fuel diameter which strikes a compromise between reduced fuel costs and reduced capital costs. For this reason, a single value for fabrication cost and the resulting fuel cycle cost cannot be used for power reactors of the same type even when the elements are superficially similar.

FIGURE 3
FUEL COSTS FOR BOILING WATER
AND
PRESSURIZED WATER REACTOR POWER PLANT

The lower limit of the band in Figure 3 represents data supplied by a leading manufacturer of boiling water reactors for Zircaloy clad, slightly enriched UO_2 fuel elements and is based upon their quoted price for fabrication and estimated burnup (for plants greater than about 20,000 kwe, burnup for the equilibrium core is estimated at 12,000 mwd/short ton).

The upper limit of the band is based upon fuel cycle costs contained in Ref. 3. These costs are based on stainless steel clad, slightly enriched UO_2 elements. Part of the reason for the higher fuel cycle costs is the smaller fuel pin diameter used in the pressurized water reactor designs.

C. Operating and Maintenance Costs

The principal components of operating and maintenance costs are labor (wages, salaries, overhead) and supplies. Often "third party liability" insurance is also included as an operating and maintenance expense.

Personnel requirements for both bwr and ocmr plants should be slightly higher than for pwr plants. In the case of the bwr, it is estimated that 1 or 2 additional operating and maintenance personnel per shift are required for a 40,000 kwe power plant to handle the complications associated with the radioactive steam cycle and the relatively large volumes of radioactive waste from the "full-flow" steam condensate demineralizer regeneration step. A similar number of additional personnel would be required for a comparable size ocmr to handle organic purification and makeup operations and the attendant waste disposition.

It should be noted that no nuclear power plant in the United States up to the present time has been operated only to produce power at the lowest possible cost. Without exception, operation of power reactors has had other goals as well, particularly "research and development" and operator training. For this reason, the number and type of personnel required to operate a plant on a "power only" basis is somewhat speculative.

In addition, labor costs are subject to very wide fluctuations depending upon:

- Operating philosophy
- Design features of the plant
- Local labor rates
- Site conditions which might affect operating procedures such as meteorology, hydrology, geology, remoteness, etc.

For example; the annual cost for an Okinawan operator, at the present time, is approximately \$1,000 (Ref. 30); for military personnel outside the United States, the annual cost is \$7,500. Estimates of the number of operating and maintenance personnel required for a 40,000 kwe nuclear power plant range from 25 to 60. Thus the estimated annual operating and maintenance labor costs for a 40,000 kwe nuclear power plant can range from \$25,000 to \$450,000 or 0.09 to 1.6 mills/kwhr (based on 80% plant operating factor). Therefore, the operating and maintenance labor cost must be evaluated on a site-by-site basis, using the number of personnel and cost per man appropriate to the particular situation.

The lack of extensive experience plus the wide variation in site conditions and plant design philosophy leads to a substantial uncertainty as to the annual cost of operating and maintenance supplies. One convenient method for estimating supplies is as a fraction of the total plant cost. Estimates for operating and maintenance supplies vary from 0.5% to 1.0% of the total plant cost per year. For the 40,000 kwe example above, at \$430/kw this would amount to 0.3 to 0.6 mills/kwhr or \$85,000 to \$170,000/year.

Since the plants considered in this study will be U. S. Government owned, the cost of "third party liability" insurance will not be included.

SECTION V

CRITERIA FOR SPECIAL APPLICATIONS

The various military applications for which nuclear power plants are being considered fall into four categories. These are:

1. Land based plants at major military bases at which other generating facilities are or will be in operation.
2. Land based plants at small isolated bases with no other source of power.
3. "One trip" barge mounted plants.
4. Mobile barge mounted power plants.

The criteria for selection and design of land based nuclear power plants at major military bases shall be similar to that which would be appropriate for a small isolated utility in the U. S. In those cases in which nuclear fuel costs are less than the cost of conventional fuel, it can be assumed that the nuclear power plant will be base loaded. Since other generating facilities will be present, a reasonable amount of shutdown for refueling and maintenance can be tolerated as long as the downtime is not in excess of approximately 10%.

For the nuclear power plants at small isolated bases, at which the proposed installation will be the only source of power supply, slightly different criteria will apply. In most cases, these plants will have a "firm power" requirement equal to or somewhat less than normal power demand. In these cases, it will be necessary to install stand-by capacity using either nuclear or fossil fuel so the firm power requirement can be met with any single unit shut down. Since the capital cost of nuclear power plants is usually considerably higher than a fossil fuel plant, the most economical type of stand-by unit to install is a fossil fuel plant. In the size range below approximately 10 mw, these plants would be diesel-generator units.

"One-trip" barge plants can be defined as a power plant mounted on a barge or floating platform which would be towed to a coastal site at which it would be more or less permanently fixed. This type of installation would be considered for those coastal sites where local construction costs are in excess of those in the U. S. Since these barge mounted plants are a substitute for an equivalent land based plant, the criteria for selection of the nuclear power plant will be generally the same as for a land based plant. However, the necessity

for mounting the plant on a sea going barge requires special attention in the design to achieve a compact plant with a low center of gravity.

Mobile barge mounted power plants are units which will be a part of the Navy's mobile power reserve and will be completely self-contained. All support facilities including quarters for the operating crew, fuel storage, etc. will be provided on the barge. The power plant should be capable of quick and reliable response to fluctuating loads, and have the greatest availability practicable since the barge may be the only source of power. The mobile barge plants will be capable of being towed from site to site after the reactor has been operated at power for substantial periods. Consequently, the plant should be designed to withstand the shock loads and "pitch and roll" likely to be encountered during the towing period without dismantling the reactor core.

SECTION VI

APPENDIX

APPENDIX AREVIEW OF WATER-COOLED AND MODERATED REACTORS -- DIRECT CYCLEConcept Description and Discussion of Advantages-Disadvantages

Direct cycle water reactors, sufficiently developed for the purposes of this study, are cooled by boiling water within the reactor core, and the steam generated is used directly to drive the turbine. Thus, the reactor is also the steam generator, and the reactor coolant operating pressure equals the saturation pressure of the steam generated. An example of a direct cycle boiling water reactor power plant is the planned Pacific Gas & Electric Humboldt Bay nuclear power station where steam will be generated in the reactor core at 1,010 psig and 547.6° F and fed dry and saturated to the steam turbine at 1,000 psig and 546.4° F (under full load conditions).

Heat removal from a boiling water reactor (bwr) core can be accomplished either with natural circulation or forced circulation, the latter being employed where it is economically desirable to achieve higher power densities. In large plants an indirect cycle is often used in combination with the direct boiling water cycle in order to improve load following characteristics as well as increase the power density -- particularly where this increased power density can be achieved as cheaply in such a "dual cycle" arrangement as in a forced circulation single cycle plant (see Ref. 17). An example of a dual cycle plant is the Dresden Nuclear Power Station rated at 180,000 kwe net. Steam is generated in the reactor core at 1,000 psig; water removed from this steam in the primary steam drum is used to generate supplementary steam in a secondary cycle at 495 psig. Dual cycle arrangements as well as forced circulation cooling would not be economically warranted for boiling water reactor applications in the power range prescribed for this study.

The bwr concept has certain inherent advantages over the previously developed pwr because of potential savings from:

1. Generating the steam within the reactor vessel, thus eliminating the heat exchangers required to transfer heat from the primary to a secondary system.
2. Utilizing the latent heat of evaporation of water.
3. Driving the turbine directly with the steam generated in the reactor increases the thermal efficiency for a given reactor system pressure.

4. Eliminating expensive canned rotor pumps in the case of natural circulation reactors, the type of bwr most applicable to the study criteria.

In addition to the above, for a given steam pressure at the turbine throttle, the operating pressure in the reactor vessel and auxiliaries is much lower in a direct cycle boiling water reactor than in an indirect cycle pressurized water reactor. Pressurized water reactors operating at 2,000 psig normally deliver dry, saturated steam to the turbine throttle at approximately 550 psig, while a boiling water direct cycle reactor is capable of delivering 550 psig steam at reactor operating pressures of less than 600 psig.

In the early stages of the bwr development there was concern with such problems as the effect of steam voids on stability, reactivity and safety, the use of radioactive steam in the turbine, and the ability of the reactor to follow load changes. Although subsequent experiments and operating experience with boiling water reactors have effectively solved most of these problems, the necessity for an external "automatic control" system to follow rapid load changes, and the limited number of fuel cladding failures which can be tolerated without plant shutdown are inherent disadvantages of direct cycle boiling water reactors.

Technological Status

1. Plants Completed, Under Construction, or Being Planned

The rapid development of the bwr concept is shown by Table A-1, page A-8, which lists all the various bwr plants in operation or being built. As a result of the experience gained from the Borax experiments and the successful operation of the Experimental Boiling Water Reactor (EBWR) and the Vallecitos Boiling Water Reactor (VBWR), boiling water power plants within the 5 to 40 mwe range can take advantage of the simplest boiling cycle, i.e., direct-cycle, natural circulation with internal steam separation.

2. Research and Development

Although the first bwr experiment was made only six years ago, an adequate technology exists today to proceed with bwr design of all sizes. Also, much of the technology available from pwr plants and research type water reactors is also applicable to the bwr concept. A brief summary of the status of these developments follows.

- a. Reactor Physics

Although the physics of a bwr is complicated by the presence of steam voids, sufficient data are available

to allow complete design of direct cycle natural circulation boiling water reactors in the size range of interest to this study.

b. Fuel Elements

Fuel element development for both boiling and pressurized reactors was directed initially toward the use of alloys of uranium metal. There are two problems associated with the use of metallic uranium fuel in central station power reactors:

- (1) Dimensional instability at long exposures.
- (2) Extremely high reaction rates with water at high temperatures, thus creating a hazard should the cladding develop leaks.

Combinations of alloying metals and heat treatment have resolved one or the other of these problems. No completely satisfactory alloy has been developed which resolves both problems simultaneously during the useful life of the elements without requiring the use of uneconomically high concentrations of alloying materials.

The difficulty with metallic uranium has led to the emphasis on the use of uranium dioxide (UO_2) fuel in water cooled reactors. This material is usually in the form of small cylindrical pellets, manufactured by compacting and sintering UO_2 powder. These pellets are then encased in Zircaloy or stainless steel tubing. Uranium dioxide does not exhibit dimensional instability after long exposures in a reactor, nor does it react with water at appreciable rates. Although there are some drawbacks to the use of uranium dioxide -- lower density, lower thermal conductivity, and higher cost than uranium metal -- it is a material which can be used to fabricate a reliable fuel element. Consequently, UO_2 is being almost universally adopted for water cooled power reactors using enriched uranium.

The cladding for uranium dioxide fuel can be Zircaloy, stainless steel, or -- at lower temperatures -- aluminum alloys. Compared with Zircaloy, the use of stainless steel results in lower fabrication cost but higher fuel material cost because of the higher neutron absorption by

steel. Aluminum alloys are limited to lower temperatures (about 400° F) than either Zircaloy or stainless steel with resulting poorer thermal efficiency and higher turbine cost.

c. Heat Transfer and Fluid Flow

Most of the research and development to date on heat transfer and fluid flow has been centered on natural circulation boiling water reactors. It is becoming clearer that, just as in the case of pressurized water reactors, boiling reactors' performance is limited at the center of the fuel to 5,000° F, the melting point of UO₂, and by a maximum surface heat flux set by "boiling burnout".

Present fuel fabrication economic studies indicate that a UO₂ fuel pellet size of approximately 0.4 to 0.5 inch OD should be used. A typical Zircaloy-2 clad, 0.45 inch OD pellet could operate at a maximum heat flux of approximately 400,000 Btu/hr/sq ft with a center line temperature of 4,900° F, based on $k = 1.0 \text{ Btu/hr/ft/}^{\circ}\text{F}$. A typical maximum-to-average heat flux of 3.5 results in an average heat flux of 115,000 Btu/hr/sq ft.

A representative design using a water-to-fuel ratio of approximately 2.5 results in an average power density of 34 kw/liter of core and an average specific power of approximately 14 mw/ton of UO₂. The velocity of the steam from the core poses another design limitation on whether effective moisture separation can be performed within the reactor vessel. From EBWR experience, the limiting value of steam velocity is approximately 1.75 ft/sec.

d. Reactor Safety

Boiling water reactors have a built-in safety characteristic, because of the formation of steam voids which reduce reactivity. This unique feature of boiling water reactors has been well demonstrated by the Borax experiments.

e. Reactor Stability

Boiling water reactors have shown excellent stability over a wide range of power densities with relatively large amounts of reactivity compensated by steam voids. Borax-IV held approximately 6.9% reactivity in steam voids at 322 psig and achieved a power density of 45 kw

per liter of core. Increases in operating pressure to 600 and 1,000 psi have shown stable operation at progressively higher power levels. Present boiling water reactor designs use operating pressures of 1,000 psi or higher. The use of UO_2 fuel has led to a much longer thermal time constant in the fuel. This, in turn, has led to greater stability of the reactor with respect to transients. In boiling water reactors, with long time constant elements, it is probable that hydraulic instability will determine the maximum power. In-pile and out-of-pile tests indicate this hydraulic instability occurs at approximately 70% exit voids from a channel.

f. Components and Auxiliaries

In general, boiling water reactors have been able to use components and auxiliaries already developed for conventional steam plants or for non-boiling water reactors. Specific developments have been required because of water activity, water decomposition, and corrosion of materials in saturated steam with high oxygen content.

g. Turbines for Use with Direct Cycle Steam

Corrosive and erosive effects of wet, oxygenated steam have been slight in the EBWR and VBWR turbines, leading to increased confidence in the use of steam directly from the reactor. In the design of turbines for bwr plants, several ground rules have been adopted to avoid problems:

Moisture of steam is held within acceptable limits by use of throw-out stages and moisture separators.

Stage energy and pressure drop are held to conservative values.

Leakage paths subject to "wire drawing" are eliminated wherever possible. Where they cannot be eliminated, they are faced with erosion resistant materials.

Connections are provided for introducing decontamination solutions for turbine cleanup.

Economics

Capital and unit power costs for natural circulation direct cycle bwr power plants in the 5 to 40 mwe range are discussed in Section IV of this report.

Figure 1, page 12, illustrates the capital costs for bwr power plants as a function of capacity. These capital costs are based on assumed 1959 U. S. construction economics excluding land purchase and interest during construction.

Figure 3, page 16, illustrates fuel cycle costs for the bwr and pwr concepts in power plant capacities in the range of interest to this study. These unit fuel costs assume a waiver on fuel-use charges and a plutonium credit at \$12 per gram.

Future Potential

Additional experience in the construction and operation of bwr plants should result in substantial improvements in power density, stability, cycle efficiency and over-all cost of power.

Appreciable reductions in unit capital costs may result from the operation of the bwr at higher than the presently specified 1,000 psi, thus improving steam temperature and pressure conditions and plant thermal efficiency. Even greater improvements might be obtained from incorporating nuclear superheating as an integral portion of the bwr reactor. If both higher operating pressures and nuclear superheating can be developed economically, the bwr would provide steam at conditions similar to conventional steam boilers resulting in savings in the turbine-generator portion of the plant and greater thermal efficiency.

Further capital savings might result from the successful attainment of "controlled recirculation" currently being developed in the Northern States Power Company Pathfinder Project and/or of the "moderator level control" technique being developed by American-Standard in their Variable Moderator Reactor research. These developments should simplify or even eliminate present mechanical control systems.

Cost reductions should result from improved plant layout studies and structural technique developments. For example, the so-called "vapor suppression" containment scheme proposed for the Pacific Gas & Electric Humboldt Bay plant may reduce plant containment costs for water reactors.

Fuel cycle costs should be reduced by continued improvements in fuel designs and fabrication techniques, by standardization and volume production of fuel types applicable to the bwr, and by increasing the average irradiation of fuel. The use of UO_2 and other ceramic-type fuel, and the reduction of peak-to-average burnup by improved programming of fuel movement or repositioning in the core should permit fuel burnups of more than 20,000 mwd per ton. With these improvements and reductions in the cost of zirconium cladding, fuel costs may be lowered to less than 3 mills per kwhr for plants in the 40 mwe size.

At this stage of bwr development and experience it is difficult to assess quantitatively the magnitude of these possible cost improvements. These developments could result in more than a 30% reduction in unit power costs for bwr nuclear power plants in the small or intermediate size range.

TABLE A-1
BWR POWER PLANTS

<u>Plant</u>	<u>Status</u>	<u>Type</u> ⁽¹⁾	<u>Net kwe</u>	<u>Startup</u>
Borax-2,3,4	Operating	A (Experiment)	2,000	1954-1958
EBWR	Operating	A	4,500	1956
VBWR	Operating	A	5,000	1957
ALPR	Operating	A	200	1958
Dresden	Operating	D	180	1959
Borax-5	Being Built	B (Experiment)	2,000	1958
RWE (Germany)	Being Built	C	15,000	1960-61
Elk River	Being Built	C	15 + 7 (coal superheater)	1960-61
Ulyanovsk (USSR)	Being Built	G	50	1961-(Est.)
Sverdlovsk (USSR)	Being Built	F	94	1961-(Est.)
SENN	Being Built	D	150	1963
JPDR (Japan)	Planned	-	11,700	1962
Humboldt Bay	Planned	A	50-60	1962
Suisaton (Switzerland)	Planned	-	15	1962
Northern States	Planned	E	62	1962
Big Rock Point, Michigan Lake	Planned	G	50	1962

- (1) BWR Types:
- A - Direct cycle, natural circulation
 - B - Direct cycle, natural circulation with nuclear superheat.
 - C - Indirect natural circulation.
 - D - Direct dual cycle forced circulation with external steam separation.
 - E - Direct, controlled circulation with integral nuclear superheat.
 - F - Graphite moderated, pressure-tube-type boiling H₂O plus integral superheat.
 - G - Direct forced circulation.

References: 4, 5 and 6

APPENDIX B

REVIEW OF WATER-COOLED AND MODERATED REACTORS -- INDIRECT CYCLE

Concept Description and Discussion of Advantages-Disadvantages

The general class of water cooled and moderated reactors utilizing an indirect cycle includes both pressurized reactors, which either completely suppress boiling, allow subcooled boiling, or allow localized bulk boiling, and closed-cycle boiling reactors. This section is concerned primarily with pressurized water reactors which have the greatest accumulated experience, and therefore constitute the most important category of indirect cycle water cooled reactors. Furthermore, only light water coolant will be considered here, since heavy water cooled and moderated reactors are considered in Appendix D.

The pressurized water reactor (pwr) utilizes light water as the moderator-coolant, with the water maintained under sufficient pressure to prevent net steam generation in the primary coolant system. The primary coolant absorbs heat in the reactor core and delivers it to a shell-and-tube heat exchanger where, depending on the primary coolant pressure, 300 to 600 psi dry saturated steam is produced in a secondary cycle. The steam thus formed drives a turbine-generator unit. The condensed steam is then processed as in conventional power plants and returned to the heat exchanger for continuation of the steam producing cycle. The pwr is thus characterized by having two separate closed loop systems: the reactor coolant loop and the steam cycle.

Depending on the required plant power range or degree of reliability and flexibility desired, either single or multiple primary coolant loops may be used. Each loop contains a steam generator and a recirculating pump.

The steam system is not radioactive barring the remote possibility of simultaneous rupture of fuel elements in the reactor core and tube leakage in the heat exchanger. This eliminates the necessity of shielding in the turbine-generator plant, and permits conventional equipment and maintenance procedures in the secondary system.

The reactor and primary coolant loops including steam generators are usually located within a containment shell to provide against accidental release of fission products to the plant environs.

In a direct cycle plant, either the turbine-generator facilities must be accommodated within the containment shell or part of the radioactive steam system must be located outside the shell.

Another advantage of the pwr over most direct cycle boiling water reactors is a relatively smaller reactor coolant cleanup system. In direct cycle boiling water reactors, it is thought necessary to provide full-flow steam condensate demineralization in order to prevent radioactivity buildup in the coolant-steam cycle and reduce biological shielding requirements.

A principal advantage of the pwr is the excellent load following characteristics resulting from a combination of the large negative temperature coefficient inherent in water-moderated reactors and the non-boiling coolant loop feature of the pwr. The experience of Shippingport (PWR) has demonstrated that this plant is highly responsive to load changes and has excellent load peaking characteristics. Normal power changes in the PWR are being met without movement of control rods.

The inherent economic disadvantage of the indirect cycle reactors, in comparison with direct cycle reactors, is the additional cost of the heat exchanger, primary coolant pumps, pressurizer, and piping required to transfer heat from the primary to the secondary system. In addition, the pwr has greater wall thicknesses for the reactor vessel and primary system due to its higher operating pressures.

Technological Status

1. Plants Completed, Under Construction, or Being Planned

As a consequence of the early and significant effort in developing and proving pressurized water reactors for naval and army applications, this reactor type has the advantage of having a proved technology and operating experience (Ref. 2). This fact is best summarized by Table B-1, page B7. Among the pressurized water reactor plants for electric power generation now or previously in operation are:

Army Package Power Reactor, SM-1 (APPR-1)

Rated Power: 2 mwe

Critical on April 8, 1957

Has produced 91,980 mwt-hrs as of April 4, 1959

Shippingport Atomic Power Station

Rated Power: 60 mwe

Critical on December 2, 1957

Has produced 334,970 mwe-hrs as of June 1, 1959

In addition, 10 to 12 pressurized water reactors for the propulsion of naval vessels are in operation. Approximately 60 naval reactors are either operating, under construction, or planned. Standardization of these reactor designs has resulted from orders for 27 S5W plants and 12 CLW-A1W-A2W plants, putting pwr propulsion reactors almost on a production basis.

2. Research and Development

The pressurized water reactor concept, because of its advanced status and successful operation, has well-defined areas in which research and development emphasis should be placed for improvement in the over-all economics, general reliability, and safety of the concept. Recently published data indicate the extent and type of research and development previously considered or currently in progress for improvement of the pressurized water reactor system (Ref. 2). The following presents a resume of the categories and general nature of this work.

a. Reactor Physics

Investigation of pwr reactor physics includes: reactor statics, nuclear design, reactor dynamics and safety analysis. Work on reactor statics is being directed toward better methods for predicting nuclear behavior of the reactors. Work on nuclear design is being directed toward improved fuel cycle performance such as flux flattening, multi-region cores, etc. The work on reactor dynamics is being directed toward a better understanding of transient behavior.

b. Fuel Elements

While slightly enriched uranium dioxide has proved a satisfactory fuel material, investigations are being conducted to determine the ultimate capability of UO_2 to retain fission gases, dimensional stability at high burnups, and optimization of fuel rod size that may be governed by the center melting temperature, burn-out heat flux, and economics.

Continued investigation of cladding materials for pwr applications offers considerable potential for the improvement of fuel economics and increased reliability at high fuel burnups. The use of zirconium alloy or stainless steel cladding currently predominates in pwr fuel designs. Stainless steels are relatively cheaper, have better structural strength and chemical compatibility with the water system. While the zircalloys involve a higher initial expenditure at this time, their much lower neutron absorption cross-section, and therefore greater neutron economy, offer a definite area wherein the over-all fuel costs can be improved significantly. Work is currently in progress to determine more economical production and fabrication of zirconium and its alloys.

c. Control Rod Materials

The investigation of materials having a high neutron absorption cross section for use in control rods has been extensive. Hafnium has been the material most used in control rods, but its relatively high cost motivates investigation of other metals or alloys. Among the materials currently undergoing either experimentation, or active consideration for use in control rods, are boron carbide, rare earths, and silver-indium-cadmium and silver-indium-cadmium-tin alloys. Use of these materials involves determination of the corrosion, stability, and chemical compatibility within the specific system, as well as neutron absorption characteristics.

d. Other Reactor Component Materials

The use of materials less expensive than stainless steel for primary system components would result not only in lower initial material costs, but also in lower installation and maintenance costs. Investigations in this direction are being conducted to determine radiation damage and erosion-corrosion rates. In addition, experimentation is required to determine the effect of integrated fast fluxes on low alloy steels, and to perfect the metallurgical and chemical requirements for more reliable installation techniques such as welding.

e. Additional Programs

Investigations are continuing in the fields of heat transfer and fluid flow measurements and calculations, and to simplify instrumentation and control systems.

Economics

Capital and unit power costs for pwr nuclear power plants in the 5 to 40 mwe range are discussed in Section IV of this report.

Figure 2, page 13, illustrates the capital costs for pwr nuclear power plants as a function of capacity. These capital costs are based on assumed 1959 U. S. construction economics, excluding land purchase and interest during construction. The spread in the capital costs for each given plant size may be partly explained by varying status of designs, design philosophy, manufacturing techniques and commercial practices.

Figure 3, page 16, illustrates fuel cycle costs for the bwr and pwr concepts in power plant capacities in the range of interest to this study. These unit fuel costs assume a waiver of fuel-use charges and a plutonium credit of \$12 per gram.

Future Potential

Three major areas appear to offer considerable opportunity for improvements in the economics of indirect cycle water cooled reactors: materials and construction techniques, plant simplification, and improved plant performance. Not all of these areas are mutually compatible. Careful design compromises will be required to achieve the most economical plant for a given size.

1. Materials

Stainless steel, which until now has been the principal construction material, may be replaced eventually by less expensive low alloy steels for use in vessels, piping, etc. It is likely that the cost of Zircaloy cladding will be reduced, thus decreasing fuel cycle costs.

2. Plant Simplification

Examples of plant simplifications are:

- a. Reducing the number of primary loops
- b. Cheaper containment methods
- c. Development of natural circulation heat removal
- d. Development of controlled leakage pumps
- e. Use of laminated pressure vessels
- f. Elimination of external pressurizers; for example, use of the reactor vessel steam dome as the primary system pressurizer.

With respect to cheaper containment methods, the "vapor suppression" technique referred to in Appendix A might also improve the economics of pwr containment. Another possibility was mentioned in the recent study of an advanced pressurized water reactor in which containment is provided by a concrete enclosure with a thin steel liner (Ref. 7). Safety valves would permit the release of the first rush of steam, but close in time to prevent release of any radioactive gases resulting from a core meltdown.

Development of natural circulation heat removal for indirect cycle water cooled plants would offer a considerable simplification by eliminating the need for expensive canned rotor pumps, coolant check valves, associated electrical power and control equipment. Such a program has already been initiated for naval vessel propulsion plants. This would be most applicable to small or medium-sized plants, such as are being considered in this study.

Improved Plant Performance

The use of multi-region, multi-cycle cores offers two improvements in plant performance. First, extension of the average exposure level of discharged fuel for the same maximum burnup, and second, considerable flux flattening.

It may be possible to reduce some of the hot channel and hot spot factors applied to present designs and still maintain safety. The application of statistical analysis of hot channel factors may be useful.

It is possible to design indirect cycle water cooled reactors to allow for a predetermined number of fuel element failures. This would take advantage of one of the inherent advantages of the pwr concept, i.e., operability with a number of failed elements without necessarily shutting down the plant.

The recent Advanced Pressurized Water Reactor Study showed the possibility of achieving a remarkably high over-all plant net thermal efficiency (Ref. 7). The APWR plant would generate 1,035 psig saturated steam in the heat exchanger. This, coupled with a reheat turbine cycle would result in a 34.5% net plant efficiency. This is accomplished by allowing net steam generation in certain areas of the core. The mixed-mean coolant temperature leaving the vessel is below the saturation temperature. This demonstrates that indirect cycle pressurized and boiling water reactors may eventually approach each other in design characteristics.

Another area of improved performance is the production of superheated steam either by a fossil-fueled superheater or by nuclear superheating in the reactor. This would decrease the cost of the secondary plant and increase thermal efficiency.

TABLE B-1
PWR POWER PLANTS

<u>Plant</u>	<u>Status</u>	<u>No. of Loops</u>	<u>Net kwe (unless otherwise noted)</u>	<u>Startup</u>
SM-1 (APPR-1)	Operating	1	1,850	1957
PWR-1 (Shippingport)	Operating	(3 out of 4 operating)	60,000	1957
PM-2A (Greenland)	Being Built	1	1,500	1960
SM-1A (APPR-1A)	Being Built	-	1,700	1960
			+ 38.3 x 10 ⁶ Btu/hr space heat	
NMSR (NS Savannah)	Being Built	2	22,000 SHP	1960
BR-3 (Belgium)	Being Built	2	11,500	1960
Yankee	Being Built	4	110,000	1960
Indian Point (1)	Being Built	4	255,000	1961
PWR-2 (Shippingport modified)	Planned	4	140,000	-
AEC Design Study (APWR)	Planned	4	236,000	-
SM-2	Planned	1	7,000	-
Saxton	Planned	1	5,000	1961
Sundance (PM-1)	Planned	1	1,000	1961
Project SELNI	Planned	4	160,000	1963

(1) Reactor thermal power 585 mwt, fossil fired superheater 210 mwt.

APPENDIX C

REVIEW OF ORGANIC COOLED AND MODERATED REACTORS

Concept Description and Discussion of Advantages - Disadvantages

In one respect the organic cooled and moderated reactor (ocmr) is similar to the pressurized water reactor concept described in Appendix B. in that both have closed primary coolant recirculation loops which transfer heat from the reactor core to a tube-and-shell heat exchanger in which steam is generated in a secondary cycle for turbine drive. The ocmr uses a suitable organic fluid instead of water as coolant-moderator. As in the pwr, bulk boiling is suppressed in the coolant, although advanced versions of both concepts contemplate some local or subcooled boiling.

The ocmr has many auxiliary systems typical to water cooled and moderated reactors such as degasification, pressurization, coolant bypass purification, coolant makeup, and waste disposal.

The organic fluid currently being employed or considered for organic reactor operations is "Santowax R" or slight variations of this commercial mixture, which consists of 1% diphenyl and 99% o, p, m-terphenyls. These hydrocarbons are good moderators, i.e. (the approximate) moderating ratio for the Santowax R mixture is 67 compared to 69 for H₂O and 21,000 for D₂O. They have considerably lower vapor pressures than water, thereby permitting operations at lower pressure than in the case of nonboiling aqueous reactors. For example, the coolant operating pressure in the Organic Moderated Reactor Experiment is about 200 psig with the coolant temperature in the reactor core ranging from 500 to over 700° F; in a pwr operating at 2,000 psig, the outlet coolant temperature is 500-550° F. Thus the lower operating pressures of the ocmr permits a considerable reduction in the required wall thicknesses for vessels, piping, valves, and other components.

Corrosion rates between the organic coolant-moderator and low alloy or low carbon steels are sufficiently small to permit the use of these cheaper materials in the fabrication of the reactor vessel and internals, piping, heat exchanger and other components in the ocmr primary system. Many components are similar to equipment developed for the refinery industry such as "hot oil" pumps. The use of these conventional materials and equipment enables certain economies in the construction of the ocmr not currently permissible in water reactor designs. Further, because of the chemical compatibility of the organic fluid with materials of construction, there is a wide choice of fuel and cladding materials possible without

potentially hazardous exothermic chemical reactions.

It is practicable to generate superheated steam with the ocmr so that higher thermal efficiency can be realized. For example, it is currently planned that the City of Piqua organic reactor power plant will produce steam at 550° F and 435 psig, and more advanced ocmr designs contemplate a generation of 650° F steam at pressures of 600 - 1,000 psig.

One final advantage of the ocmr concept is that, because of the lower corrosion rates and the absence of oxygen atoms in the coolant, the radioactivity level of the circulated primary coolant is considerably lower than for water-cooled reactors.

The ocmr has limitations, as well as problems and uncertainties that must be resolved before this reactor can be considered sufficiently reliable (both with respect to technology and economics) for the remote military power applications required before 1965. Probably the principal disadvantage of the ocmr for the purposes of this study is the lack of operating experience required to establish sufficient definitive design and operating criteria to permit a reasonably accurate economic assessment of this concept. Fuel elements proposed for the ocmr are only now being tested in the OMRE. Further, the fact that the OMRE is fueled with SM-1 type fuel (stainless steel-clad fully enriched UO₂ dispersed in stainless matrix), creates doubts with respect to the planned use of slightly enriched fuel. For example, experience has been gained in OMRE operations to permit some prediction of the degree of radiolytic degradation of the organic fluid. However, the ratio of fast neutrons to gamma in the radiation from slightly enriched fuel will be appreciably greater than currently being experienced with the highly enriched fuel. Thus, there is a question of whether or not the extent of organic degradation with attendant cleanup and makeup requirements can be predicted on the basis of ionizing radiation per se or on the basis of the type of radiation. The amount of organic fluid which would have to be purified and replenished could appreciably affect the economics of a small or intermediate size plant in remote areas.

Inherent limitations in the ocmr concept include:

The temperature limitation due to the combined pyrolytic and radiolytic effect on the organic fluid with resultant deleterious effects on its viscosity, heat transfer, and other properties.

Inherently poor heat transfer characteristics of organic fluids thus requiring extended surface fuel elements with attendant increases in fuel fabrication costs.

An appreciable volume change of the organic coolant (34% between 200 and 700° F) necessitating provisions for a large expansion tank to take care of temperature transients.

The need for tracing the primary system with either electric or steam heaters to melt the organic at startup since the terphenyls and other organics considered for ocmr application are solid at room temperatures.

Flammability of the organic coolant requiring fire protection provisions similar to those employed in the petroleum industry. For example, p-terphenyl has a fire point of 460° F. However, its auto-ignition point is about 1,100° F and since the upper operating temperature of the organic coolant will be below this, spontaneous ignition in the event of leakage is not considered likely.

Technological Status

The concept of utilizing an organic material as the moderator and/or coolant for a nuclear reactor was first considered during the Manhattan Project. Because of the unknown radiation damage characteristics and because of the relative unavailability of enriched uranium, the concept did not receive much further consideration until about 1950. At that time, a program of basic research was undertaken via cyclotron irradiations, in-pile capsules, in-pile loops, and heat transfer loops to determine the extent of decomposition of the organic coolant under irradiation and to determine the heat transfer characteristics.

Design work on the Organic Moderated Reactor Experiment (OMRE) was started in the summer of 1955. Construction of this reactor was initiated in mid-1956 and was completed in the Fall of 1957. The OMRE has been operating since early 1958, with a considerable backlog of operating experience having been accumulated to date.

The principal objectives were to obtain plant operating experience, to establish the actual decomposition rate of the organic coolant in an operating reactor, and to determine the heat removal characteristics of the coolant with various amounts of "high boiler" derivatives present. In this respect, the decomposition of the organic coolant has been shown to be quite comparable to that determined by earlier experimental work.

The OMRE utilizes highly enriched uranium oxide dispersed and clad in stainless steel. The fuel elements for the initial experiment were similar to those used in the SM-1 at Fort Belvoir. These elements are not the same as would be used in a power reactor.

Some prototype power reactor elements have been inserted in the OMRE and these are discussed later.

The Piqua organic power reactor will be operated by the City of Piqua Municipal Power Commission, as a part of the U. S. AEC's Second Round Power Demonstration Reactor program. The reactor plant will supply steam to the Piqua Power Station where 11.4 mwe will be generated. The plant is scheduled for operation in 1961.

The fuel for the Piqua reactor will be slightly enriched metallic uranium alloy, clad in aluminum. The presently proposed fuel element design employs two concentric hollow cylinder-type fuel elements with helically spiraled fins. The fuel alloy composition for the present design is uranium alloyed with 3.5 wt% molybdenum.

Design studies have been performed for small-sized organic reactors in the sizes 10, 20 and 40 mwe (Ref. 12). All of these designs employ metallic uranium alloy fuel elements clad with aluminum. Fuel element designs are similar to those planned for the Piqua reactor.

Recently, Atomics International and Bechtel completed conceptual design studies for advanced organic moderated reactors with capacities of 75 and 300 mwe. These studies employed uranium oxide pellet-type fuel elements clad in Aluminum Powder Metal (APM). The core design was based also on the utilization of sub-cooled or nucleate boiling in certain regions of the core.

In addition to the above studies for power reactors, an ocmr plant was proposed for use in propelling a T-7 oil tanker.

Studies have also been made combining the use of organic coolant with other moderator materials such as heavy water, graphite, beryllium, zirconium hydride and light water. These studies were performed by Atomics International, Brookhaven National Laboratory, Canadian-General Electric and others.

At the present time an advanced organic moderated reactor program is being carried out under the sponsorship of the U. S. AEC. This work involves research and development on physics, fuel and material properties, heat transfer and fluid flow, coolant chemistry, reactor safety, component and auxiliary systems, and fuel handling.

The two major areas of research and development effort necessary to prove out the advanced ocmr power reactor concept are investigations on the use of APM-clad UO_2 or uranium carbide fuel and the use of local boiling in the core. Recent studies conclude that a net total of \$7,000,000 - \$8,000,000 of research and development cost will be required (Ref. 8 and 11).

In addition to the above research and development work, an Experimental Organic Cooled Reactor (EOCR) is being built as a supplement to the present OMRE (Ref. 4 and 9). The EOCR will be a 30 - 40 mwt organic moderated and cooled experimental reactor providing space for the in-pile testing of full-sized prototype power reactor fuel elements and various organic coolants.

One of the major uncertainties in the technological status of the ocmr concept is the lack of a proven fuel element. Although the highly enriched uranium oxide-stainless steel SM-1 elements used as the OMRE fueling can be considered as a demonstrated fuel element, the use of this type of fuel element imposes a considerable penalty in fuel costs.

The fuel elements proposed for the Piqua plant and in the preliminary designs of the 10, 20 and 40 mwe ocmr power plants consist of metallic uranium alloy clad with aluminum. Unfortunately, uranium alloy fuels are limited in burn-up by a swelling of the fuel caused by the expansion of fission gases within the metal lattice (see Figure C-1). There is little evidence among available data to show that significant restraint to this swelling can be obtained by various alloying techniques so that even the best alloys developed to date will not allow exposures or temperatures to as desirable a level as will uranium oxide. Thus, recent interests in the development of suitable ocmr fuel has been directed to the use of uranium oxide clad with a special sintered aluminum alloy containing aluminum oxide (APM or SAP). To date no prototype fuel elements of this design have been adequately tested.

With reference to investigations of the proposed Piqua fuel, two flat plate assemblies containing 2S aluminum-clad uranium-based alloys with small additions of Mo + Si and Mo + Al were irradiated in the OMRE to a peak burn-up of about 3,000 mwd/t at which time cladding failures were experienced. These failures were attributed to the presence of particulate matter in the organic fluid being circulated which caused the plugging of coolant channels resulting in hot spots with consequent thermal degradation of organic coolant and fuel cladding. Three additional elements are now being tested in the OMRE --- one of a circular tube design. At present these tests are still incomplete.

Economics

The capital cost for ocmr designs for the size range of interest in this study varies from \$800/kw at 10 mwe to \$430/kw at 40 mwe (Ref. 12). Utilizing a fixed charge rate of 5%, the capital costs amount to 5.7 mills/kwhr for the 10 mwe size and 3.1 mills/kwhr for the 40 mwe size. The capital costs are based on U. S. construction, 1959 cost levels, and no interest during construction.

Fuel cycle costs range from 5.6 mills/kwhr at 10 mwe to 4.8 mills/kwhr at 40 mwe, based on the waiver of fuel use charges and a Pu credit of \$12/gm.

The estimated cost of make-up of organic coolant due to nuclear degradation is approximately 0.7 mills/kwhr.

Section IV-C discusses operating and maintenance costs for an ocmr power plant.

Since interest is not included in the fixed charges used in this study a plant, such as the ocmr with relatively low capital costs but high fuel and operating costs is at an economic disadvantage compared with a plant with high capital costs but low operating costs.

Future Potential

As implied above, a major advancement in the organic reactor concept would be realized in the development of a proven fuel element design which would permit higher temperatures, much greater burnups than presently permissible with metallic fuel elements, and appreciably lower costs. If concurrent with this development, the heat transfer capabilities of the system could be improved -- for example, in the utilization of local boiling in the core -- then substantial increases in the power density of the ocmr with attendant higher thermal efficiencies and lower capital costs would be achieved. Further improvements in the ocmr would be realized in reducing the degradation rates of the organic coolant.

The major potential in the development of an improved fuel element for the ocmr exists at present with the proposed uranium oxide clad with the sintered aluminum metal-aluminum oxide product. Another promising possible fuel material is uranium carbide in lieu of uranium oxide.

Further improvement in heat transfer capability, as well as in the reduction of organic degradation may be realized in the development of alternate organic fluids or additives to the present mixtures. Extensive investigations and experimentation are in progress with regard to this search. Organic substitutes being studied include xylene and silane compounds. The development of aluminum alloys or the aforementioned sintered aluminum-aluminum oxide product for structural purposes in the ocmr core would make possible a significant improvement in the neutron economy of the reactor.

It is predicted that unit power cost savings up to 40% of those estimated for plants using current technology might be obtained through realization of the above potentialities.

FIG. C-1

VOLUME CHANGE VS. IRRADIATION TEMPERATURE

APPENDIX D

REVIEW OF HEAVY WATER MODERATED REACTORS

Concept Description and Discussion of Advantages-Disadvantages

Heavy water as a moderator for thermal (neutron energy) reactors offers the advantages of a low neutron absorption cross section plus fairly good moderating power. This combination permits the use of natural uranium fuel. With a heavy water (D_2O) moderator, D_2O can be used as the coolant, if desired, in both pressurized and boiling reactors. Other coolants, including light water, organic materials, liquid metals, and gases can also be used.

As with light water reactors, heavy water moderators exhibit a large negative temperature coefficient of reactivity with its attendant safety and controllability. In fact, heavy water reactors have an additional safety feature which is a longer neutron lifetime than light water reactors.

The penalty associated with heavy water moderated reactors is in its neutron slowing down properties. The slowing down length (distance the fission neutrons must travel before reaching thermal equilibrium) is quite large and results in a fuel element lattice spacing on the order of 10 or 11 inches for optimum design. If the heat output per fuel element is limited (e.g., by boiling burn-out), the physical size of a heavy water reactor for a given output is larger than a corresponding light water reactor. At power levels greater than about 200 mw, the reactor core then becomes so large that pressure vessels for pressurized D_2O or boiling D_2O reactors become impractical. This limitation can be bypassed, however, by going to a pressure tube concept. The penalty associated with this concept is due to the loss in neutron economy as a result of absorption in the material added to the reactor core by the pressure tube material. On the other hand, by insulating the pressure tubes and maintaining the moderator at a low temperature, the fission cross section (which decreases with increased moderator temperature) is improved, the amount of reactivity required to override the moderator temperature coefficient of reactivity is reduced and a greater fuel element lifetime can be achieved.

The pressure tube concept also allows the use of coolants other than D_2O . Light water, an organic fluid or a gas might be used to reduce the required D_2O inventory and loss due to leakage or to provide a coolant with better heat transfer properties. The most important advantage of the pressure tube concept is that there essentially is no nuclear system limit to the size of these reactors so that for large power requirements, the predicted unit power costs become quite attractive.

Technological Status

At the present time, there are no D₂O moderated power reactors built or in operation in the United States except for the aqueous homogeneous reactor (HRE-2). Even so, there is a considerable amount of applicable experience available from the D₂O moderated production reactors at Savannah River and from the Argonne and Chalk River research reactors as well as from the construction of the Plutonium Recycle Test Reactor (PRTR) at Hanford and the Heavy Water Components Test Reactor (HWCTR) at Savannah River. It may also be expected that experience will be available from the boiling heavy water process heat reactor presently operating in Norway.

Along with a great number of studies on a variety of types of D₂O moderated reactors, a considerable amount of research and development is being conducted on materials, fuel elements, pressure tubes, pressure vessel structural problems, flow problems, physics and control.

Two D₂O moderated power reactors have been proposed to be built in the United States by 1963. The first will be a power demonstration reactor for the Carolinas-Virginia Nuclear Power Associates Inc. (CVNPA) at Parr, South Carolina. Westinghouse as the principal contractor will provide the design for a 17 mwe, slightly enriched, D₂O cooled, pressure tube (U-tube design) reactor. Although the fuel will be enriched to about 2%, the reactor is designed as a prototype of a large plant which could operate on natural uranium.

The second plant presently planned in the United States will be built in Florida for the East Central and Florida West Coast Nuclear Groups with General Nuclear Engineering Corporation (Combustion Engineering) as the nuclear design agent. This plant will be a gas cooled, 50 mwe, pressure-tube design using slightly enriched uranium. The full scale plant of the same type would be capable of operating with natural uranium fuel.

In Canada, the D₂O cooled 20 mwe "NPD-2" reactor is under construction at Des Joachims, Ontario. The "NPD-2" is a prototype for CANDU, a 200 mwe natural uranium fueled reactor planned for operation in late 1964 or early 1965.

A small 9 mwe pressurized D₂O (pressure vessel type) reactor is under construction in Sweden with startup due in 1960.

To date, there has been no serious attempt to develop D₂O moderated reactors for small power plants in the 5 to 40 mwe range except as a prototype of a larger plant or a test reactor.

The Experimental Boiling Water Reactor (EBWR) was designed for future operation with heavy water as the moderator coolant; however, actual operation with D₂O has been deferred.

Economics

The only cost data on D₂O moderated reactors in the size range of interest to this present study is on the CVNPA prototype. The capital cost of the plant is predicted to be about \$23,000,000 excluding research and development, fuel inventory and fabrication, substation and transmission lines.

A preliminary design of a 69 mwe prototype boiling heavy water nuclear power plant, produced by Sargent & Lundy and Nuclear Development Corporation of America, predicts a capital cost of \$720/kw (Ref. 15 and 16). It is interesting to note that the capital cost of the D₂O inventory accounted for 20% of the total.

These figures are considerably higher than those for light water reactors of the same size. Moreover, the small capacity plants do not permit achieving the potentially lower fuel costs which D₂O offers, compared to light water.

Future Potential

The programs now active both in the United States and other countries are likely to result in reduced capital costs for heavy water reactors, at least in plants of 100 mwe capacity or larger. However, their inherent large physical size and complexity compared with light water reactors will always result in higher capital costs within the 5 to 40 mwe range (especially if the heavy water inventory is included in the capital cost). In this low power range, it is unlikely that the combined fuel and moderator makeup costs for heavy water reactors will be sufficiently less (if less at all) than fuel costs for light water reactors to offset the higher capital costs.

APPENDIX E

REVIEW OF GAS COOLED REACTORS

Concept Description and Discussion of Advantages/Disadvantages

Gas cooled reactors can be defined simply as a nuclear power reactor in which all or most of the fission heat is removed from the core by passing a gas over the fuel material. Air, nitrogen, helium, carbon dioxide, hydrogen, and helium-neon mixtures have been used or proposed as coolants in gas cooled power reactors. Because of their low density, the gases at practical pressures will exhibit negligible neutron moderation and it is necessary to use another material as a moderator. Graphite has been used as a moderator in gas cooled power reactors in most applications. Other moderators which have been used or proposed include heavy water, beryllia and light water.

If temperatures greater than approximately 1,200° F are achieved in a gas cooled reactor, power can be generated in a direct cycle by passing the gas through a closed-cycle gas turbine. A special case of a direct cycle plant would be use of an open cycle in which air would be used as the reactor coolant as well as the working medium in the turbine. This particular concept is being investigated for aircraft propulsion.

For temperatures below 1,200° F, it is usually more economical to use an indirect cycle; i.e. using the hot gas to generate steam which, in turn, passes through a steam turbine.

Gaseous coolants can achieve high temperatures at modest pressures, less than 1,000 psi. In addition, gases such as helium are non-reactive with the materials of construction. As a result, it is possible to achieve a higher coolant outlet temperature with gas cooled reactors than is practical with water or organic coolants. This high temperature operation results in good thermal efficiency, and in the case of an indirect cycle, reactor steam conditions which approach those achieved in modern fossil-fuel fired steam plants. However, it has not been possible to take advantage of the high temperature capability of the gaseous coolants because of temperature limitations inherent in the fuel materials presently used and fuel element jacketing materials.

Another advantage of gaseous coolants is the low macroscopic neutron cross section of the coolants which results partially from its low density and, in the case of helium and carbon dioxide, the low microscopic neutron cross sections.

The principal disadvantage in gas cooled reactors to date has been poor power density which has resulted in large physical size.

Technological Status

Until recently, there has been no significant development of gas cooled reactors for power generation in the United States, and there are no reactors of this type presently in operation in this country. In Great Britain, on the other hand, there has been a substantial interest in this type of reactor; eight gas cooled power reactors are presently in operation at Calder Hall and Chapel Cross. In France two reactors (G-2 and G-3) are presently operating at Marcoule.

In both Great Britain and France the reactors are carbon dioxide cooled, natural uranium fueled, graphite moderated, indirect cycle units. The units now operating have a capacity per reactor equivalent to 30 to 40 mwe net. As a result of the successful operation of these units, much larger plants are presently being built in Great Britain with generating capacities up to 250 mwe using a single reactor. In addition, two high temperature reactors, i.e., the AGR and Dragon projects, are presently planned in Great Britain.

Within the past two years, extensive study in the United States has resulted in the AEC's present program to develop gas cooled reactors for central station power production. This program is centered around the construction of three plants, the Experimental Gas Cooled Reactor (EGCR), scheduled for completion 1962, the High Cooled Reactor (HWGCR), both scheduled for completion in 1963, (Ref. 4).

The EGCR and HTGCR, in common with the British and French reactors, will be graphite moderated, indirect cycle units. However, they will take advantage of the availability in the United States of partially enriched uranium as the fuel and helium as the coolant. These two modifications should result in lower capital costs and higher remperature steam with higher thermal efficiency.

Tables E-1, E-2 and E-3, pages E4, E5 and E6, summarize some of the features of the EGCR, the HTGCR and the HWGCR, respectively. The capital costs of each of the plants are estimated at \$25,000,000 to \$30,000,000. In addition, the research and development program associated with the HTGCR is estimated at over \$14,000,000; the program associated with the HWGCR is estimated at \$15,000,000.

In addition to the above, three additional programs are developing gas cooled reactors for special applications: The Army Gas Cooled Reactor Program, the Maritime Gas Cooled Reactor Program and the direct cycle Aircraft Nuclear Propulsion Program. All three programs are directed towards direct cycle plants using gas turbines as the prime mover.

The Army Gas Cooled Reactor Program is directed towards the development of a mobile power plant with the capacity of approximately 400 kw using an enriched uranium fueled, light water moderated, nitrogen cooled reactor, coupled with a closed cycle gas turbine. The first of two steps in the development of this power plant is the Gas Cooled Reactor Experiment (GCRE), located at the National Reactor Testing Station in Idaho began operation early 1960. This unit, which has no power conversion equipment associated with it, cost approximately \$3,800,000. A parallel effort at Fort Belvoir, Virginia, is directed towards the Gas Turbine Test Facility (GTTF) in which a closed cycle turbine is being developed using an oil fired heat source. The second step of this development will culminate in the construction of the ML-1 nuclear power plant. This will combine a gas cooled reactor similar to the GCRE with a closed cycle gas turbine similar to the GTTF.

Both the Atomic Energy Commission and the Maritime Commission are interested in developing a direct cycle gas cooled reactor for propulsion of merchant ships. The present program is directed towards the ultimate construction of a Maritime Gas Cooled Reactor Experiment with completion of this experiment planned for 1963. In addition to the problems inherent in developing an economical high temperature gas cooled reactor suitable for propelling merchant ships, a parallel effort must be devoted towards developing a suitable closed cycle gas turbine of the required size, 20,000 to 30,000 shaft horsepower.

The direct cycle gas cooled reactor is one of the principal approaches towards achieving a nuclear propelled piloted aircraft. Much of the information on the design characteristics of the reactors being developed for aircraft propulsion remain classified. However, it is known that the problems in developing the extremely high temperatures and high power output per unit weight have not as yet been resolved. While it is possible that much of the information developed in the course of this program would be applicable to reactors for electric power generation, no substantial effort is being devoted to such a program.

TABLE E-1

TECHNICAL ASPECTS OF THE EXPERIMENTAL GAS COOLED REACTOR (EGCR)

Thermal Output	84.3 mw
Electrical Output (net)	25,700 kw
Fuel	UO ₂ Pellets
Enrichment	2.55% Steady State Feed
Loading	12,800 kg of UO ₂
Cladding	Stainless Steel Type 304
Moderator	Graphite
Coolant	Helium
Reactor Outlet Temperature	1,050° F
Reactor Pressure (inlet)	315 psia
Steam Conditions	1,300 psia, 900° F, 254,000 lb/hr
Experimental Facilities	4 in-pile through loops 9.5" diameter
	4 in-pile through loops 5" diameter

TABLE E-2

TECHNICAL ASPECTS OF THE HIGH TEMPERATURE GAS COOLED REACTOR (HTGCR)

	<u>First Core Metal-Clad Graphite Elements</u>	<u>Future Core Graphite-Clad Elements</u>
Thermal Output	92 mw	115 mw
Electrical Output (net)	28,500 kw	40,000 kw
Fuel	U-Th carbide dispersed in graphite	Same as metal clad
Enriched U-loading (93% U-235)	190 kg U-235	190 kg U-235
Thorium loading	894 kg	1,190 kg
Cladding	Stainless steel	Graphite
Moderator	Graphite	Graphite
Coolant	Helium	Helium
Reactor Outlet Temperature	1,015° F	1,382° F
Reactor Pressure	300 psia	300 psia
Turbine Steam Conditions	850 psig, 850° F, 296,000 lb/hr	1,450 psig, 1,000° F, 367,000 lb/hr

TABLE E-3TECHNICAL ASPECTS OF THE HEAVY WATER GAS COOLED REACTOR (HWGCR)

Thermal Output	150 mw approx.
Electrical Output	50,000 kw
Fuel	UO ₂ Pellets
Enrichment	1.8% U-235
Loading	15,800 kg UO ₂
Cladding	Stainless Steel
Moderator	Heavy Water
Coolant	CO ₂
Reactor Outlet Temperature	1,050° F
Reactor Inlet Pressure	500 psig

Economics

Analysis of the economics of gas cooled reactors of interest to this study is limited to the indirect cycle designs, both graphite and D₂O moderated, represented by the EGCR, HTGCR, and HWGCR. The direct cycle plants either have not been adequately developed as in the Maritime Gas Cooled Reactor and Aircraft Nuclear Propulsion Programs, or are not available in the sizes of interest to this study.

The cost of a 26,100 kw power prototype similar to the EGCR at Oak Ridge but without some of the experimental features has been estimated to be approximately \$28,100,000, or \$1,100/kw. For plants of this type fuel costs are estimated to be approximately 2.7 mills/kwhr.

The cost of the 28,500 kw HTGCR is estimated to be \$24,500,000, or \$860/kw. Fuel costs for the HTGCR are not available.

The cost of the 50,000 kw HWGCR has been estimated at approximately \$570/kw. Although fuel costs for the HWGCR are not available, they should be about the same as those for the EGCR type plant. Any savings possible because of the less neutron absorbing moderator would be offset by D₂O losses.

The large physical size and greater complexity will probably lead to somewhat higher operating and maintenance costs for indirect cycle gas cooled reactors in this size range compared with water cooled reactors. For example, the estimated operating and maintenance cost for the 26,000 kw EGCR type prototype is 4.7 mills/kwhr. A comparable figure for a boiling water reactor is approximately 2.5 mills/kwhr.

Future Potential

The large physical size and complexity of indirect cycle gas cooled reactors makes them attractive only in large capacity units. It is unlikely that in the 5,000 to 40,000 kwe range, indirect cycle gas cooled reactors will become competitive with the water or organic reactors.

The simpler direct cycle gas cooled reactor might be attractive in this size range if developed specifically for power production. Successful operation of the Army Reactors Program's ML-1, construction and operation of the Maritime Gas Cooled Reactor experiment, and/or achievement of higher-than-design temperatures in the indirect cycle plants would give a substantial impetus to the development of medium sized, direct cycle gas cooled reactor power plants.

SECTION VII

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B. Glossary

abs	- absolute
APPR	- Army Package Power Reactor
bbl	- barrel - 42 U. S. gallons
Borax-1, II, III, IV, V	- Boiling Reactor Experiment - Nos. 1-5
BR-3	- Belgian Reactor No. 3 (pressurized water)
Btu	- British thermal unit(s)
bwr	- boiling water reactor
EBR-1	- Experimental Breeder Reactor-1
EBWR	- Experimental Boiling Water Reactor
°F	- degrees Fahrenheit
ft	- foot, feet
g	- gram(s)
gpm	- gallons per minute
hr	- hour(s)
kg	- kilogram
kw	- kilowatt(s)
kwhr	- kilowatt hour(s)
mwd	- megawatt days
mwe	- electrical megawatt(s)
mwT	- thermal megawatt(s)
ocmr	- organic cooled and moderated reactor
psi	- pounds per square inch
pwr	- pressurized water reactor
PWR	- Shippingport Pressurized Water Reactor Power Plant
SM-1	- Stationary Medium power reactor - No. 1 (operating at Fort Belvoir, Va.)
SM-2	- Stationary Medium power reactor - No. 2
sq ft	- square foot, feet
SRE	- Sodium Reactor Experiment
t	- ton
UO ₂	- uranium oxide
VBWR	- Vallecitos Boiling Water Reactor