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THE FEASIBILITY OF AN UNATTENDED NUCLEAR POWER PLANT

M. W. Rosenthal
M. M. Yarosh
C. F. Baes, Jr.
R. C. Robertson

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Abstract

A study has been made of the feasibility of constructing a small nuclear power plant capable of operating one year completely unattended. A system which will produce 1 Mw of electricity without interruption is required for field use in about four years.

It is the opinion of the authors that the four-year requirement greatly restricts the opportunity for development of new concepts and new technology and that the objective is most likely to be attained by extreme simplification of a type of reactor with which there has been favorable experience. A pressurized-water reactor was selected for the application, both because of the extensive experience with it and because it appears readily adaptable to simplification. For example, the power plant suggested has hermetically-sealed primary and secondary systems, and all control systems, except the turbine governor, have been eliminated.

The authors conclude that it is feasible to develop a reliable, simplified, pressurized-water reactor system for unattended service in the allotted time.

INTRODUCTION

A study was undertaken by staff members of the Oak Ridge National Laboratory to evaulate the feasibility of constructing a small nuclear power plant capable of operating unattended for one year. The objectives of the study were, specifically, (1) to assess the potential for success of a program to design and develop such a system within a limited time period and (2) to propose, if possible, concepts which offer promise of meeting the design objectives. The singularly most important consideration was the degree of unattended reliability achievable from the individual components of a nuclear plant and from the combined operating system.

In estimating the potential for reliability of a nuclear power plant, it was necessary to conceive of a design capable of satisfying the system requirements. The primary objective, however, was estimation of the potential for success of a program based on such a design, rather than presentation of the design itself.

As will be evident throughout the report, the design considerations have been restricted to meeting only the stated objectives of the study. It was the opinion of the authors that the achievement of one year of unattended operation was a problem of such difficulty as to dominate all other design considerations. Thus the question, what makes a plant cease to function, was continually in the forefront throughout the study. That this apparent obsession with reliability is a requisite may be attested to by the experience records of those reactors which are operating today.

For the present study the following system requirements were established:

1. The plant is to operate unattended at full power continuously for a minimum of one year.

2. The net electrical output is to be 1000 kw of three-phase alternating current at 4160 v.

3. The frequency can be selected by the designer within a range of 60 to 1000 cycles, but the frequency selected must be held to within ±1% throughout the life of the plant.

4. The reactor plant will be the only power source connected to its electrical load.

5. A reactor power plant capable of satisfying the preceding requirements is to be in operation on location within approximately four years.

6. Interruption of the generation of electricity for any reason within one year of initiation of operation is considered a plant failure.

7. Short-term (one-year) reliability is of utmost importance and will not be sacrificed to provide long-term life.

8. The system could be considered expendable, if necessary, at the end of one year of operation.

9. During startup and shakedown of equipment, semiremote operation of the plant will be possible, and special equipment may be provided for startup requirements.

10. After the beginning of unattended operation, no communication with the system will be possible other than knowledge that the electrical load is being supplied.

11. Within reason, plant size and weight are not considerations.

12. Within reason, cost is not a consideration.

13. Plant efficiency is not an important consideration.

14. The system will operate stationary.

15. The plant will not operate in a populous area.

16. The sink for heat rejection will be determined by the application, but adequate means of cooling will be available.

The time limitations imposed on this study precluded an extensive examination of all systems which might conceivably satisfy the requirements. Hence, only those systems that appeared to offer the maximum potential for reliability were studied. There is no intent to imply that the systems discussed in this report comprise the only ones capable of satisfying the plant requirements, but, within the framework of reliability considerations for this system, they appear to be the most promising.

The authors are deeply indebted to various members of the Oak Ridge National Laboratory and the Oak Ridge Gaseous Diffusion Plant staffs for numerous contributions to this study. L. D. Schaffer helped apprise the authors of the status of small-reactor development. E. R. Mann advised the authors on reactor control. W. C. Thurber and P. Patriarca were consulted on metallurgical problems and the selection of fuel elements. E. J. Breeding and W. G. Cobb were consulted regarding bearings and seals, J. L. Gabbard regarding electrical generators, and P. H. Pitkanen regarding core physics.

Helpful discussions were held with W. B. Cottrell, J. E. Cunningham, A. P. Fraas, C. H. Gabbard, E. E. Gross, C. J. Hochanadel, P. G. Lafyatis, H. C. McCurdy, H. F. McDuffie, A. M. Perry, I. Spiewak, E. Vincens, C. S. Walker, and members of the Engineering Development Department, Technical Division, Oak Ridge Gaseous Diffusion Plant. Burns and Roe, Inc., consulting engineers to the Laboratory, studied certain problems of the

secondary system and contributed information pertinent to component selection and reliability.

During the course of the study, discussions and correspondence were carried on with personnel of a number of organizations. Many of the opinions expressed in this report are a result of such discussions, and the authors gratefully acknowledge the following organizations for their valuable contributions:

Aerojet General Corporation (GCRE)

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Allis-Chalmers Manufacturing Company

Atomics International

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General Electric Company (Erie, Pennsylvania; Fitchburg, Massachusetts; and Schenectady, New York)

Gilbert Associates

International Nickel Company

Martin Company (Baltimore, Maryland)

Personnel of SM-1 and GTTF, Fort Belvoir, Virginia

Pierce Governor Company

Westinghouse Electric Corporation (Bettis Plant, East Pittsburgh, and Lester, Pennsylvania)

Woodward Governor Company

Worthington Corporation (Harrison, New Jersey)

Gilbert Associates kindly made available to the authors a draft of their forthcoming report[1] on the reliability of reactor components.

SOME COMMENTS ON THE ACHIEVEMENT OF RELIABILITY

In considering the objectives of this study, it is meaningless to speak of building a reactor system which will operate without failure for one year. To do so would suggest we are requiring it to be a certainty that a particular reactor power plant operate successfully for that period. Actually, a more realistic specification might be that we expect nine out of ten reactors to be operating at the end of a year, which is equivalent to requiring that an individual reactor have a reliability of 0.9. For purposes of this study, it was assumed that a plant reliability of 0.9 was required for one year of unattended operation.

Some understanding of what is involved in achieving a 0.9 reliability may be obtained by considering how the reliability of the individual components affects the over-all reliability of the power plant. If a system consisting of a number of components is to have a reliability of 0.9, the reliability of each component in the system must be better than 0.9 and an average component reliability very much better than 0.9 may be necessary. This is illustrated by Eq. (1) in which p_i is the probability that component "i" will last one year, and P is the probability that all the components will last one year:

$$P = p_1 p_2 p_3 \cdot \cdot \cdot p_i \quad . \tag{1}$$

For example, if the system consists of 10 components, each having a reliability of 0.99, the probability of the system lasting one year without failure of any of the components will barely exceed 0.90. If there are 50 components, the failure of any one of which will stop the functioning of the system, the "average" reliability of each component must be 0.998. The significance of a 0.99 requirement may be more clearly understood if one pictures a test system in which 100 identical components are simultaneously set into operation. For a 0.99 reliability, 99 of the 100 components must still be operating at the end of a year.

Actually, the problem is more complicated than just assessing the probability that a particular reactor can operate unattended for one year. No nuclear power plant having the needed capacity has been built to operate without regular maintenance. For reactors such as the SM-1 (APPR) and the SL-1 (ALPR), it is an accomplishment to operate 1000 hr without shutdown, even with maintenance operations being performed during that period. Hence, it is not a question of whether an existing system can meet the specifications, but whether a modification of an existing system or a new design of reactor can be made sufficiently reliable for unattended operation. The problem is to estimate the likelihood that a reactor system having the necessary reliability can be developed in about four years.

In this study we have attempted to distinguish between (1) the probability that the development problems associated with unproven concepts or untested equipment can be solved in a limited time period, and (2) the probability that a particular system based on tested equipment and proven concepts can operate for a year unattended. For convenience, we shall designate the first probability as P_1 and the second as P_2.

One can conceive of systems which, because of inherent features, appear to be capable of long periods of unattended operation (have a high value of P_2) after some important development problems associated with them are solved. An example might be a natural-convection liquid-fuel reactor that uses thermoelectric devices for the generation of electricity. There would be no moving parts to wear and no control system to fail. A power plant of this type, if developed successfully, would probably operate unattended for long periods. However, there would be an appreciable risk (low P_1) in proceeding to develop it for attainment of the objectives, since some of the development problems might still be unsolved at the end of four years.

On the other hand, a reactor very similar to the SM-1 reactor might be used. If no major changes were made in the control system, water-treatment system, etc., one could be confident that a reactor of this type could be built within the time allowed and that it would operate at its design capacity, but it would not operate very long without attention. This reactor concept might be said to have a high P_1 value but a low P_2 value. Since it is the product of P_1 and P_2 that gives the probability of successfully developing a system of the required reliability in the time allotted, close attention must be given to each in selecting a program to follow.

The language of mathematical probability may help to clarify the problem, but the difficulty of evaluating the probabilities remains. The uncertainty in P_1 is illustrated by several reactor development programs which have not attained their objectives after a number of years of effort. While solutions may eventually be found to their problems, a requirement of success in a limited time would not have been met. Estimation of P_2 is also uncertain, since it requires knowledge of the reliability of a number of components. Although the reliability of a component which has been extensively tested may be known with some assurance, even a small change in design can alter the value importantly. Thus little confidence would be placed in an estimate of the reliability of a complex piece of equipment not yet designed.

In order to assess the reliability of existing equipment, operation reports on a few reactors (SM-1, SL-1, PWR) were perused in order to determine the frequency and causes of failures, and operating experience was discussed with personnel at several installations (SM-1, SL-1, GCRE, GTTF). A study was also made of the operation history of a number of moderate-sized steam turbines.

The reactors examined were not meant to operate unattended, and they were designed for regular maintenance. Hence, it was expected that these systems, having hundreds of mechanical components and large numbers of electronic devices, would be far less reliable than required for one year of unattended service. More surprising and sobering was the conclusion that few existing components in normal use would operate for one year without maintenance. This conclusion is least applicable to the mechanical equipment in the primary system of pressurized-water reactors, because intensive effort in the Navy program has produced reliable components. It applies very strongly, however, to steam power equipment, where spares are provided for many items, and appreciable on-stream maintenance is the normal practice. Electronic control devices that use vacuum tubes are also quite unreliable for one year of service. An extreme illustration is provided by the SM-1 control system, in which 300 vacuum tube replacements were made during the first year of operation.[2]

The data on turbine-generators were of particular interest because they indicate how reliability can be improved. From a study of the 14-year operating history of a number of conventional 25-Mw turbine-generator units, Myers[3] concluded that only five units in 100 would be capable of operating continuously for one year. Myers' study was extended to three 3000-kw turbine-generator units, and a value of 0.05 for the reliability was again obtained. Further examination of the failures of the smaller systems indicated, however, that if the need for replacing the generator brushes and repacking the admission valves could be eliminated, approximately 60 units in 100 could operate continuously for one year.

There are three ways one might proceed to improve the reliability of a system which is to operate unattended: (1) eliminate unreliable components by eliminating their functions, (2) improve the reliability of the equipment by design and development, and (3) duplicate less reliable items so that if they fail, spares will automatically continue to perform the function.

Many components in reactor power plants appear to perform functions that are not necessary under the ground rules of this study. An example is an overload protection device, which will sometimes interrupt service unnecessarily and hence is an undesirable item with regard to achievement of uninterrupted operation. Where continuous power production is the only criterion by which performance is judged, stoppage of a pump because the circuit breaker trips is a plant failure equivalent to stoppage because the windings burn out.

The reliability of components can be improved in several ways. Modifications, such as substitution of better materials, use of better methods of fabrication, tighter quality control and inspection, and more extensive testing of an item before acceptance, may increase the probable life of an existing device. In addition, the change from normal requirements resulting from the specific objectives of the design may permit the use of a different type of device to perform the required function. As pointed out in the Gilbert Associates study[1] of reactor reliability, the achievement of reliability is not consistent with a minimum-cost philosophy.

In considering the use of spares, an important point to recognize is that improved reliability does not necessarily accompany duplication. An insufficiently reliable control device for switching to a duplicate component, for example, could reduce the over-all reliability of the system. Duplication of a component which might develop a leak doubles the chances of that type of failure.

One way to improve reliability, particularly in control equipment, is to use coincidence circuitry which requires, say, that two out of three parallel devices operate before the function is performed. With this type of circuitry, spurious operation of one device does not cause the function to be performed, and failure of one device to operate does not prevent performance of the function. It is worth noting that a system of this type works best under supervision so that a component which has failed can be repaired to return the system to its initial capability. In principle, however, the system can be made as elaborate as required to achieve the needed reliability.

It is likely that reliability requirements on individual components can be relaxed somewhat if advantage is taken of the concept of operating spares. For example, suppose that a system has two circulating pumps, both operating all the time, but each having a capacity such that one alone will provide sufficient flow for reactor operation. With this system the reactor is not prevented from continuing operation by the failure of one pump. If each of these pumps has a reliability of 0.9, then the probability, as computed from the following equation, is 0.99 that at least one of the pair will still be operating at the end of a year:

$$P_g = 1 - (1 - p)^n \quad , \qquad (2)$$

where P_g is the probability of survival of at least one component in the group, p is the probability of survival of an individual component, and n is the number of identical components. If there are three pumps, each having a 0.9 reliability, then the probability is 0.999 that at least one will be operating at the end of a year.

The time allowed for the development of a reactor system has a very strong effect on the selection of a program. As stated in the Introduction, this study presumes there is need for a reliable reactor to be in the field in about four years. If a prototype reactor is to be operated before the first field installation is made, and this seems essential, four years is a fairly short time for the program. The time required for construction of a small reactor, excluding development time, is indicated by the experience with three reactors in the 1-Mw (electrical) range:

1. The SM-1 (APPR, pressurized-water reactor built by Alco Products at Fort Belvoir) was critical 29 months after award of the contract. A conceptual design, which included a fairly detailed description of the core, was available before the contract was awarded.

2. The PM-2A (pressurized-water reactor for polar region, also built by Alco Products) was assembled in the manufacturer's plant 14 months after award of the contract. A fair amount of design work had been done before the contract date. This reactor is a skid-mounted air-transportable adaptation of SM-1.

3. The PM-1 (pressurized-water reactor being built by Martin Company for location at Sundance, Wyoming), as presently scheduled, will require about 27 months from the beginning of a parametric study to assembly of the reactor. About six months of this was required for final design, and about 15 months is estimated for the period from beginning of order placement to assembly of the system. The PM-1 is an advance from the SM-1, and it uses a new fuel element design. Fuel element development and physics studies began considerably in advance of the 27-month period referred to above.

In the last two cases, it should be noted that the times given are for assembly of the reactor. They do not allow for testing before going to power.

If it is presumed that a program is to be undertaken with some urgency and a fairly liberal budget, based on the figures for the three reactors above and on discussions with reactor manufacturers, the time required for design and construction of a prototype reactor (built for high reliability) appears to be about two years, once the design concept is established. Allowing six months for finishing the conceptual design and making arrangements for the detail design and manufacture, the prototype might be ready for testing in about two and one-half years, as shown by Fig. 1.

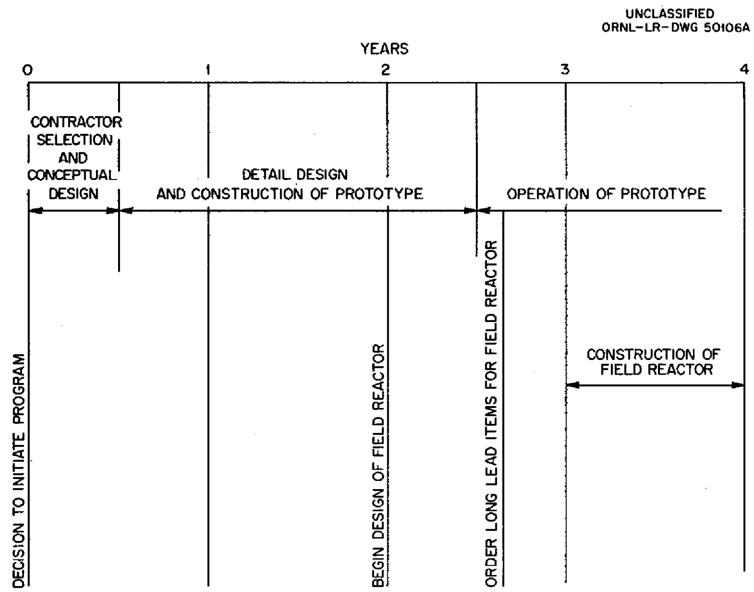

Fig. 1. Time Schedule for Achievement of Field Reactor in Four Years.

The design of the first reactor for field use might begin before completion of the prototype, and all orders for equipment for it placed after one-half year of test operation. Allowance of one year for construction of the field reactor would bring its completion time to four years from the beginning of the program.

This schedule has no provision for developmental work other than that which proceeds concurrently with the design and construction of the prototype. There is no contingency allowance for difficulties in the development of the prototype, nor is time permitted for the solution of the problems found during operation of the prototype reactor before construction of the second reactor commences. Hence, even with very little development required and with good luck at every step, about four years would be required to get the first reactor in the field. If there were delays in administrative decisions, construction holdups, or unforeseen development difficulties, well over four years might expire before a reliable power plant were achieved.

In view of the above, a development program based on the following general principles appears to have the greatest promise of success in producing a reactor to satisfy the reliability requirements in a period of about four years:

1. The main effort should be devoted to perfection of a system based on concepts with which there has been favorable experience. This restriction refers not only to reactor and power plant types in the broad sense, but also to the manner in which the systems are to be operated.

2. Approaches based on unproven concepts should be studied as secondary programs if they offer promise of giving a better system than that based on proven concepts.

3. Exceptions to points 1 and 2 could be made for parts of the system if a switch in midprogram to a proven concept would not delay the project.

4. The most promising approach to the achievement of high reliability is simplification. The objectives of this study make simplification of the reactor particularly promising, since the system need not be repairable, have a long life, follow a varying load, or operate in a populous area.

5. A close interdependence of the system parts is permissible, since in any case a failure would terminate the ability of the system to produce electricity.

6. Components should be overdesigned and underworked to increase their life expectancy. Efficiency should be sacrificed to gain reliability.

7. Components which are relatively unreliable can be used during startup if they cannot affect the system once it is in operation.

8. The individual components and the assembled system should be tested thoroughly before the system is left unattended. This testing should continue long enough to pass the period where early failures resulting from manufacturing defects are likely to occur.

The preceding principles can be summarized by saying that in seeking the achievement of reliability in the limited time available, one should attempt to use proven concepts and present technology. The success of a program to develop a reliable reactor in four years should not depend on major advancements in technology or the perfection of new concepts. If more time were available, the restriction on new development would be relaxed, but it would be stated even more strongly for a shorter time allotment.

The approach described in this report may not lead to the best system for unattended operation. Nevertheless it appears to be the approach which has the highest likelihood of producing a reliable reactor in four years.

SELECTION OF A REACTOR POWER PLANT

Reactor Type

Early in this study a comparison was made of a number of reactor concepts with regard to their applicability to an unattended system. These included the various liquid-fuel, organic-moderated, gas-cooled, and liquid-metal-cooled reactors, as well as the pressurized-water and boiling-water variations of light-water-moderated reactors. Based on the premises of the preceding section, the conclusion was reached that the extensive and generally favorable experience with water-cooled reactors recommends them for this application. The potential advantages of other systems, such as high thermal efficiency, favorable fuel or neutron economy, low construction costs, and possibly light weight, are of secondary importance in the framework of this study.

Once having selected a water-moderated reactor as the basis for this study, a further decision between boiling-water and pressurized-water systems was required. Although these reactor types have many features in common, each has certain advantages and disadvantages relative to the other. The obvious advantage of the boiling-water system is the elimination of several major items of equipment: the primary heat exchanger, the main coolant circulating pump, and the core pressurizer. An advantage of the pressurized-water reactor is that there is no production of radiolytic gas. In contrast, the boiling-water reactors in operation at present produce radiolytic gas which is either vented to the atmosphere or recombined in a catalytic recombiner. Another advantage of the pressurized-water reactor is that reactor control is accomplished very simply, and there are no problems of stability. In addition, there has been much more experience with pressurized-water than with boiling-water systems.

The heat exchanger on a pressurized-water reactor, although large and expensive, can be made quite reliable if care is taken in the selection of materials and in fabrication. Pressurizers are not of themselves basically unreliable, but the pressure control systems sometimes are. It appears to be possible to design a system in which the control, as normally performed, is eliminated, thus making the pressurizer function reliable. The most questioned component is probably the primary coolant pump. Experience with canned-rotor pumps of the type needed indicates that an individual pump that is carefully tested and found to be sound will operate reliably well in excess of one year. In addition, the concept of operating spares is directly applicable to the primary coolant pump.

When there is an overpressure of hydrogen, radiolytically dissociated water in a pressurized-water system is recombined internally with no production of hydrogen and oxygen. All pressurized-water reactors are operated in this manner. Although, in principle, water decomposition in a boiling-water reactor can be suppressed by increasing the pressure, operating at a high pH, and injecting hydrogen,[4] this mode of operation has not appeared attractive, and all boiling-water reactors in operation evolve radiolytic gas. Besides the problem of handling the radiolytic gas, the oxygen in the steam makes the materials problems more severe in the boiling-water reactor than in the secondary system of a pressurized-water power plant.

Highly enriched pressurized-water reactors can operate, and often do (the SM-1 always), without automatic reactivity regulation, except for the inherent action of the negative temperature coefficient. As discussed later, it appears to be feasible to operate a pressurized-water system for one year with no movement of absorbing materials in the core. This can be done using only proven concepts. Boiling-water reactors normally are operated with continuous reactivity regulation by movement of absorbing materials. Long-term operation without the use of an automatic control system would depend on more of an innovation than in the case of a pressurized-water reactor. (Boiling-water reactor control might be considerably simplified, however, at steady power.)

Because it appears possible to make a simple pressurized-water reactor with few innovations, the pressurized-water concept was made the basis for this study. This selection was not made because the pressurized-water reactor is inherently more reliable than the boiling-water reactor, but because there is more reason for confidence that a reliable, unattended, pressurized-water reactor can be developed in four years. There has been extensive experience with pressurized-water reactors, and the method of operation proposed here deviates little from the manner in which they have been operated in the past.

Power Recovery System

The selection of a power recovery system to operate in conjunction with the reactor plant was heavily influenced by the principles outlined in the discussion on reliability. During the initial phases of this study, consideration was given to the possibility of converting heat energy to electrical power through other than the use of a conventional turbine-generator system. In particular, a brief study was made of thermoelectric generators. The conclusion of the study was that, in view of the present early stage of development of thermoelectric materials and the relatively restricted amount of operating experience, thermoelectric generation does not appear to be promising for use at this time.[5] The development of improved materials permitting higher efficiencies at moderate temperatures may in the future make such systems attractive for unattended power stations.

With the selection of a pressurized-water reactor and application of the stated reliability concepts, it is clear that the most suitable power system is one employing a conventional steam turbine-generator set using water as the working fluid for the heat-power cycle. It is appreciated that many of the corrosion problems associated with water systems might be avoided with other fluids, but the technology and experience with equipment using less corrosive fluids are not adequate for a reliable design.

CONCEPT OF A SIMPLIFIED PRESSURIZED-WATER REACTOR

Following the selection of the pressurized-water concept a critical examination of pressurized-water reactors was required to determine the modifications necessary to achieve the required reliability. The advantages of using proven concepts and present technology were balanced against the gain to be made by a change in design and the likelihood of the change being successful. Some components and procedures used on SM-1, for example, appear to be directly applicable to an unattended reactor. In other cases, modifications are needed, and in a few instances the development of new components would be desirable. Wherever innovations are suggested in this report, there are alternate solutions which could be used should the new development not appear to be progressing rapidly enough. All the new components proposed represent combinations of items on which there has been successful experience.

In the subsections which follow, the features of a pressurized-water reactor that affect its reliability are discussed. This section does not present a reactor design, but it does suggest concepts and techniques on which a design could be based. A schematic diagram of a simplified pressurized-water reactor is presented in Fig. 2 to indicate the type of system which emerges from the discussion. All the components that function during unattended plant operation are included.

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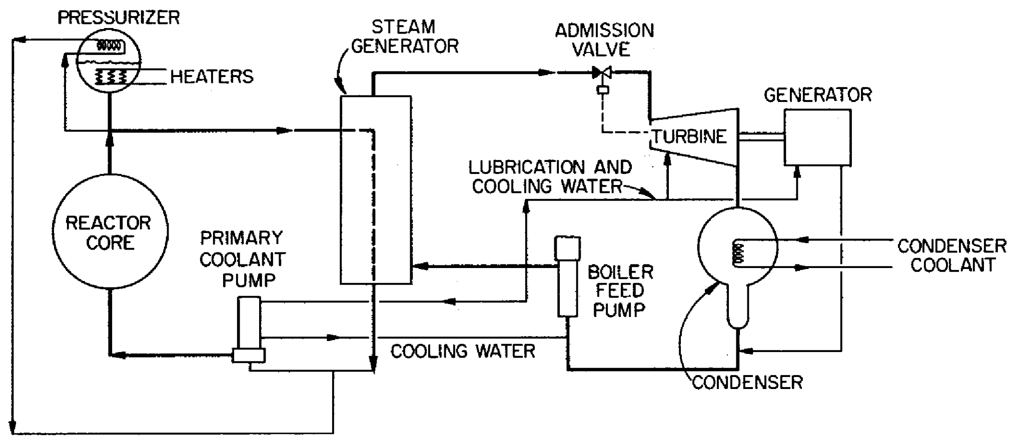

Fig. 2. Simplified Pressurized-Water Reactor With Hermetically Sealed Primary and Secondary Systems.

Primary System

The basic heat generation and removal system of a pressurized-water reactor is relatively simple, but the auxiliary equipment associated with it is often complex. The major complexities are in the reactor control system, and, in some cases, the water-treatment system. It appears possible, however, to satisfy the present control requirements with a very simple system based on the inherent self-regulation characteristics of a highly-enriched pressurized-water reactor. Water treatment does not appear to be needed in a hermetically sealed primary system to be operated for one year. The use of a hermetically sealed system with inherent nuclear control reduces the primary system to essentially the core, heat exchanger, coolant circulating pump, and pressurizer. Various features of the primary system are discussed below.

Reactor Control

The action which controls the fissioning rate in a nuclear reactor may be broken down into three functions. These are regulation, shim, and safety. In a pressurized-water reactor the regulation function maintains the primary water at a constant temperature despite small changes in reactivity, such as those associated with changes in load. Shim control performs the same type of compensation for the large reactivity changes associated with fuel depletion and accumulation of fission-product poisons. The nuclear safety system stops the chain reaction in the event of malfunction of the reactor system. Interrelated with these functions are the scheduled actions which start up and shut down the reactor.

The regulation function on a pressurized-water reactor is an inherent one. The negative temperature coefficient is of sufficient magnitude to control small changes in reactivity with only slight changes in the average core temperature. Long-term changes in reactivity during the life of a core may, however, be appreciable. The upper curve[6] of Fig. 3 illustrates the reactivity change which would occur in a fully-enriched

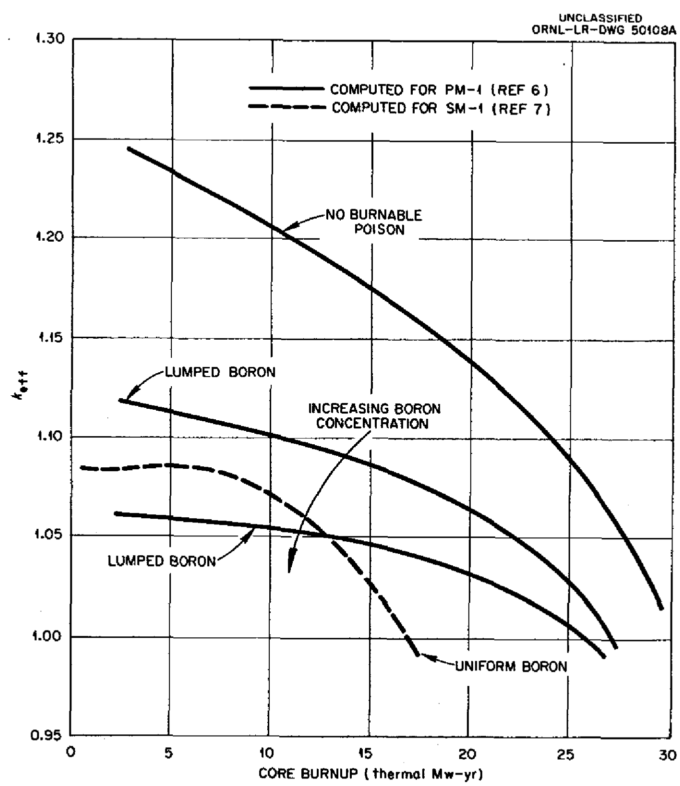

Fig. 3. Illustration of Effect of Burnable Poison on Reactivity Change of Pressurized-Water Reactor.

pressurized-water reactor if there were no compensation by the motion of a shim rod. The curve begins after the reactor has been heated from room temperature to its operating level and after the initial reactivity has been reduced by the accumulation of xenon. (The xenon poison fraction reaches about 90% of its equilibrium value in one day.)

The addition of boron as a burnable poison serves to reduce the excess reactivity during the early part of the core life, as illustrated in Fig. 3. The two lower solid curves[6] of Fig. 3 are estimates of the effects on PM-1 of different amounts of lumped boron. The dashed curve[7] shows the reactivity change with burnup in an SM-1-type core with uniformly distributed boron; the analytical model used in computing this curve accurately predicted reactivity changes during the life of Core I of SM-1. The dashed curve remains within a range of 1% $\Delta k/k$ for a core life of approximately 8 Mw-yr (thermal), and the lower solid curve remains within that range for an even longer period. One year of operation of a reactor producing 1 Mw of electricity would result in core burnup of 7 to 8 Mw-yr (thermal).

One difficulty with the use of a burnable poison is that it tends to reduce the reactivity lifetime (the period during which k_{eff} exceeds 1.0) of the core. If a shorter reactivity life of the core can be accepted, curves even flatter than those in Fig. 3 can be obtained. Other measures, such as using more than one absorbing element, closely tailoring the poison location and concentration, and perhaps even tailoring the fuel location and concentration, may extend the period in which the reactivity remains fairly constant. The uncertainties in the physics calculations, however, become greater as the system becomes more complex or as it deviates more from the region of experience.

A reactor having a temperature coefficient of -2×10^{-4} $(\Delta k/k)/°F$ would drop in temperature by 50°F to compensate for a 1% change in reactivity. (The temperature coefficient of the SM-1[8] is more negative than -2×10^{-4}.) If the burnable poison restricted the reactivity to a change from burnup of 1%, permitting the water temperature to vary over a range of 50°F would eliminate the need for mechanical control during routine operation. This technique appears to be feasible as a means of obtaining high reliability for a period of one year.

The preceding discussion has been based on the use of fully enriched fuel. Use of low-enrichment uranium might appear attractive. In a partially enriched element an excursion is curtailed by the Doppler effect in the fuel, whereas in fully enriched elements there is dependence on the change in moderator density. Actually, a strong fuel temperature coefficient is undesirable with regard to achievement of inherent control. The proven low-enrichment fuel elements for power reactors contain bulk UO_2, as discussed in the next section. The contact between UO_2 and the cladding is such that there may be an appreciable temperature drop between them, and the thermal conductivity of the gas in the gap changes significantly upon the evolution of xenon and krypton (unless the element is initially filled with xenon). In addition the effective thermal conductivity of the UO_2 may possibly change with time because of cracking and fuel burnup. The UO_2 temperature would increase during the life of the core from these effects. The uncertainty in the temperature change would make the increase in resonance absorption unpredictable, and this would make it more difficult to limit the reactivity change to a specified range. For this reason fully enriched uranium appears to be the more attractive fuel for a reactor which does not have an automatic control system.

It should be noted that the system proposed will be stable with respect to xenon "transients." A small reduction in electrical load would result in an initial decrease in neutron flux, followed by a transitional increase in xenon poisoning. The increased poisoning would lower the reactor temperature and reduce the temperature driving force for heat transfer to the secondary system. This sequence would continue if there were no corrective action. However, in a system with a fixed electrical load, the thermal power would actually increase at the lower reactor temperature, since the lowered thermal efficiency would result in a greater heat load for the same electrical power. The flux would thus tend to increase and stabilize the system.

This inherent stability may not exist if, as discussed later, the electrical load is automatically varied to control the frequency. With control obtained by varying the electrical load, a reduction in reactor temperature would be followed by a reduction in electrical load and probably a reduction in the thermal load on the reactor. While a system of this type might be stable, this can only be determined by study of the particular case.

There are three areas of concern relative to a reactor with inherent control: (1) the effect of core design on the temperature coefficient, (2) the uncertainties in judging the proper fuel and poison loading, and (3) the difficulty of accurately controlling the boron loading in a fuel element. Changes in size, shape, fuel-to-moderator ratio, reflector thickness, poison concentration, etc., may change the temperature coefficient of reactivity. In the design of a self-regulating core, achieving a favorable temperature coefficient is as important as limiting reactivity changes with time, and it should be studied as thoroughly.

The problem of estimating reactivity may, for convenience, be separated into two parts, estimation of the initial critical mass and estimation of the change in reactivity with burnup. The first part involves an attempt to make the reactivity at the beginning of unattended operation equal 1.0 at the temperature desired; the second involves estimation of changes which are in a range of about 0.01 $\Delta k/k$. A distinction is made between these two parts of the problem because the cause and effect of errors is different for each. For example, a 3% error in the initial reactivity might mean a 60°F error in water temperature if not corrected, whereas a 3% error in the change in $\Delta k/k$ would have an insignificant effect on operation of the reactor. A miscalculation of the first type is determinable, however, and probably correctable before operation of the reactor. An error of the second type would not be known until a core had been operated.

There is an appreciable body of experience with cores of the general SM-1 type that would be of value in estimating the initial reactivity, and critical experiments could be performed if needed. In estimating reactivity changes with burnup, the calculational technique used would be normalized against any experimental data that are applicable. There has been little experience in this area, however, and there are uncertainties associated with the calculations.[9] An extensive physics program is clearly called for, and it would be well if the calculations could be checked against an experiment early in the program. A particularly useful approach would be to construct a core designed for inherent control and install it in an existing reactor of the same general type. There will be several suitable small reactors in operation by the time a core could be ready for testing. In any case, physics problems are not likely to delay achievement of a reactor with inherent control, since a core with characteristics similar to those of the dashed line in Fig. 3 could be used if a core having better reactivity characteristics were not achieved.

The preceding discussion of burnable poisons has been concerned with the calculation of the amount of poison required. Another problem is that of insuring that the desired amount of burnable poison is included in the fuel element. This is discussed later under the section on fuel elements.

Even if burnable poisons are used for restricting long-term reactivity changes, some method must be provided for insuring that the core is subcritical before it is brought to the operating temperature. In addition, compensation must be made for the reactivity change which results from the initial accumulation of reactor poisons. Thus there must be a control system for bringing the reactor to the condition in which the reactivity is on the flat part of the reactivity-burnup curve. Having poison rods perform this function during the startup period would not reduce the reliability of the system, since the rods would not participate in the operation of the reactor when it is unattended. The system for providing controlled movement can be quite simple, since rapid insertion would be under the force of gravity.

There are absorber materials which have adequate reliability for use in the control rods of an unattended reactor. Unclad hafnium would satisfy the requirements, and it is likely that a matrix of europium oxide in stainless steel would suffice. (Europium elements are being tested in the SM-1 at present.) Other materials may also be satisfactory, since, being normally withdrawn, the rods would undergo little burnup.

The inclusion of poison rods would be advantageous for another reason. If late in the development program it were decided that the inherent control system were not desirable, or if the objectives for the system changed, the rods would be available for shim control with no appreciable change in reactor design. The rods could be used to change reactivity during the life of the core if a means were provided for moving them out at intervals by small amounts. For example, a watt-hour meter located on the electrical output of the turbine could actuate a simple mechanism for moving the rods. Alternatively, the rod movement could be programmed in advance for a mechanism operated by a clock. This type of control would appear to be much more reliable than one which monitors the core temperature and attempts to compensate continuously for changes, but, to operate as designed, <u>all</u> the rods would have to move when directed. If communication with the reactor were possible, the output of a thermocouple could tell an operator when to move the rods.

A fast-acting safety system does not appear to be required for a low-power-density pressurized-water reactor, particularly in a remote location. Probably the most hazardous time during the life of the reactor is the startup period. During startup the system would be under the control of an operator, and means could be provided for dropping the poison rods on either an automatic or a manual signal. Once the reactor was in operation, there would be little chance of a rapid addition of reactivity. Power removal would be steady, and there would be no pump startups or condition changes which might provide the normal source of a cold-water accident. Operator error would be eliminated along with the operator. In general, when a water-cooled reactor is operating at a high power level, it is difficult to add reactivity at a rate which would endanger the integrity of the system.

From the preceding discussion, there appears to be no need for a fast-acting safety system, except perhaps during startup. Some method would have to be provided, however, for shutting down the reactor at a signal from outside the system or at a specified time in the life of the core. In addition, rod insertion would be prescribed upon failure of the electrical output from the generator (since in any case this would signal that the reactor was not operating as designed) and possibly upon high temperature or high pressure. There is the possibility of a spurious signal causing an unnecessary shutdown, but a multiplicity of independent circuits (perhaps a three-out-of-five system) could be used to make the protective system more reliable. Aside from the automatic shutdown equipment, the system proposed would depend solely on the inherent properties of the core for controlling the reactor when it was unattended.

The problem of removing fission-product decay energy has not been considered, but the design should be such that there would be adequate coolant flow by natural convection for afterheat removal. The ultimate rejection of heat to the surroundings would depend on the specific application of the reactor.

Fuel Elements

There are satisfactory fuel elements for use at the power densities and temperature levels envisioned for a reactor of the type proposed. Core I of the SM-1, for example, appears to have performed adequately for over two and one-half years with a burnup of almost twice that required for the present application. Although SM-1 elements with full exposure have not yet been examined in a hot cell, their behavior (and examination at the reactor site) indicates that there have been no failures.

A number of cores of the same general type (i.e., flat plates of fully enriched UO_2 in a stainless steel matrix clad with stainless steel) have been built, and there is an established manufacturing capability for this design. While the SM-1 design would be quite adequate for the present application, improved elements may result from changes in fabrication methods and from modifications, such as the use of spherical oxide particles.

Elements with bulk UO_2 contained in stainless-steel tubes also appear to give reliable performance, as indicated by exhaustive tests and by experience with the PWR. Bulk UO_2 elements, however, are normally only partially enriched with U^{235}. As discussed above under the heading Reactor Control, partial enrichment is less attractive than full enrichment for a reactor with inherent control. Other types of fuel element, such as plates of Zircaloy-2-clad zirconium-uranium alloys, or the tubular stainless steel—UO_2 matrix elements developed by Martin for the PM-1, might be satisfactory. From a reliability viewpoint, however, there appears to be no necessity for use of other than the proven SM-1 type of element.

There has been a problem with the inclusion of uniformly distributed boron in SM-1-type fuel elements. During fabrication of the SM-1 core, a large fraction of the boron was lost from the fuel. Research conducted during the past two years has, however, revealed the mechanisms by which boron is lost and pointed the way to achieving better control over the final concentration. The boron loss during sintering of the stainless steel—UO_2 meat can now be closely regulated, but the loss on fabrication of the fuel plate is less controllable or predictable. At present, the final boron content of a fuel plate would probably be within 10% of the specified value. Work is continuing on this problem, and improvement in control of the boron content of the fuel element is to be expected. (For lumped poison, quite accurate control of the boron content could be achieved by adding machined strips or rods of boron-containing materials.)

The desirability of using poisons other than boron would be determined by the reactivity advantage to be obtained as balanced against uncertainties associated with their physical inclusion in the reactor. It might appear desirable to use other fuels, for example, plutonium isotopes,[10] to assist in the control of reactivity during the life of the core. The technology of including plutonium in a fuel element has not been developed nearly so far as that of uranium, however, and it is therefore advisable at present to use only uranium fuel.

Primary Coolant Pumps

As was discussed above, one advantage of a boiling-water reactor would be the elimination of the primary circulating pumps. Alternatively, a natural-circulation pressurized-water system might be used. However, it appears that experience with canned-rotor pumps of the size needed for a 1.0-Mw (electrical) reactor is sufficiently favorable that not much advantage would be gained by eliminating them, particularly if their elimination would mean going to a system with which there has been no experience. Although one may point to many examples of failures of large canned-rotor pumps, a detailed examination will show that most of the failures either occurred early in the life of the pump, as a result of manufacturing defects, or they were a consequence of malfunctioning of other parts of the system. Once a particular canned-rotor pump has been checked and found to be good, there is a high probability of its lasting a year and possibly even well beyond a year. As an example, both of the core circulating pumps of the SM-1 have operated without failure for a period of two and one-half years in which they have been subjected to many starts and stops (on one occasion a circuit breaker tripped without discernible cause).[11] In order to insure that there are no defects in design and construction of the particular pumps to be used, they should be tested in a loop and then run for a period during the checkout of the assembled reactor.

For added reliability, it would be possible to employ two continuously operating pumps in parallel, with each having sufficient capacity so that if one failed the flow would still be high enough to continue to cool the core. Some type of simple check valve would be used to avoid bypassing of the core by recirculation through an inoperative leg. Since the pressure drop would be low and a perfect seal would not be needed, it would not be difficult to design a valve to give the needed reliability. The procedure when switching from one pump to another in the SM-1 is to start the second pump, so as to have both operating, and then to cut off the first pump. There have been no difficulties associated with this method of operation.[11]

In the case of the SM-1, one pump will produce a flow of about 3900 gpm through the core, and if both pumps are operating, the flow is 4600 gpm.[11] Hence, in this system, the flow would decrease only about 15% upon the failure of one pump. A careful selection of pump characteristics and operating points may result in an even smaller flow change, although the above appears to be satisfactory.

Core Pressurizer

The pressurizer in a reactor primary system keeps the pressure high enough to prevent boiling in the core. This function should be retained in the reactor design, since there has been little experience with either local (subcooled) or bulk boiling in pressurized-water systems, and reactor control would be more difficult if boiling were permitted. In addition to preventing boiling in the system, the pressurizer normally acts as a surge tank to accommodate changes in the volume of the primary coolant. The pressurizer will have to be designed to handle appreciable volume changes in a system where water temperature variation is used for reactor control. (A 50°F temperature change causes about a 5% change in liquid volume.)

While there have been designs for self-pressurized reactors (which allow boiling near the core exit) and for pressurizers using nuclear heat, the electrically heated pressurizer should be retained because of its proven capability. The major limitation on the reliability of pressurizers in existing systems is that associated with the device which controls the pressure. Normally an instrument senses the pressure in the system and uses a control to turn the heaters off and on.

The reliability of the pressurizer system would be increased if the necessity for sensing the system pressure and for switching the heaters on and off could be eliminated. A system that requires no control or sensing of pressure has been devised to permit continuous operation of the electrical heaters. For constant conditions in the primary system, the pressurizer heaters can be carefully sized so that at equilibrium conditions the temperature of the pressurizer is at the design level.

Such a heat balance is obviously precarious in that the temperature of the pressurizer is a function both of the electrical heat input and the heat transfer coefficient. In the system described, the temperature difference between the pressurizer and ambient might be 400°F, and thus a 10% change in heat generation or heat transfer coefficient would result in a temperature change of 40°F (equivalent to perhaps 400 psi).

If the temperature drop from the pressurizer to the heat sink were small (say 50°F), the presumed 10% change in temperature differential would only amount to a few degrees. By insulating the pressurizer from the ambient and providing within the pressurizer a heat sink cooled by the exit water from the reactor core, the pressurizer-to-sink temperature differential and thus the absolute change in pressurizer temperature required to accommodate small changes in heat transfer conditions can be sharply reduced. Since it is the temperature difference between core and pressurizer that prevents boiling, the control of this difference will produce satisfactory operation.

A pressurizer design based on the above principle of maintaining the pressurizer temperature a specific amount above the core exit water temperature by using a fixed heat input would require little developmental work, and the system should operate reliably over the life of the reactor. A system of the type described is illustrated in Fig. 4. The heat input to the water in the pressurizer is provided through a number of parallel electric heaters, each operating independently of the other and separately fused, if necessary. If there were 20 independent heaters, loss of the heat output of several would not jeopardize operation of the reactor.

Steam generated by the heaters would be condensed on tubes cooled with core-outlet water. If the condenser tube wall were deliberately made thick, the heat transfer resistance could be concentrated in the wall and would not be sensitive to changes in the heat-transfer coefficient on either side of the tube. One problem of this particular design is the interference of hydrogen (which would be present in the pressurizer) with the condensation process. If this appears to be a serious problem, it may be possible to keep the hydrogen from accumulating by connecting

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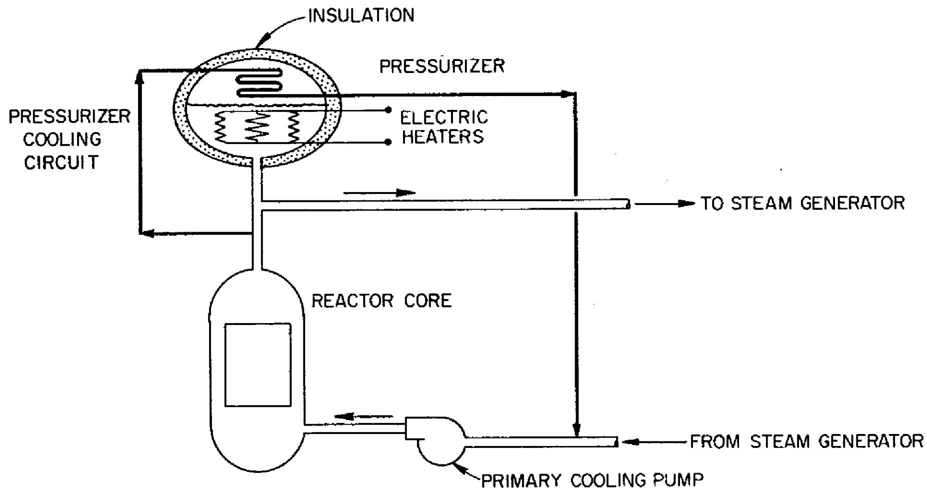

Fig. 4. A Self-Regulating Pressurizer.

a small vent line from the top of the pressurizer back to some lower-pressure point in the primary system.

There are variations of this concept which perhaps would work as well or better than the one illustrated in Fig. 4. One possibility is to provide for the vapor generated in the pressurizer to condense directly on a free surface of water from the core. Another is to locate a heat-removal coil in the liquid volume of the pressurizer rather than in the gas space. Heat transfer from the electric heaters to the coolant would be by natural circulation of the water in the pressurizer. Operation would not depend upon condensation and would not be affected by gas accumulation, but the natural-convection coefficients might be sensitive to changes in water level in the pressurizer. One advantage of having both elements in the water would be that the natural-convection coefficient decreases as the heat transfer rate falls off and thus tends to stabalize the temperature difference between the pressurizer and the primary system. In contrast, the condensing coefficient tends to increase as the heat transfer falls off and thus tends to reduce the temperature difference faster than the heat generation rate decreases.

It would appear that any of these designs could be made to work quite reliably. However, if attempts to avoid pressurizer control are not successful, an automatic control system using duplication or coincidence circuitry could be made to be reliable.

Main Heat Exchanger

Since the heat exchanger has no moving parts and no control equipment, for this unit, reliability is synonymous with leaktightness. In order to assure leaktightness, not only initially but for the desired life, the following procedure would be followed:

1. A material would be selected that is easily fabricated, has good corrosion resistance, and is not subject to stress-corrosion cracking. Inconel appears to be such a material. Its use is discussed more thoroughly under the heading Water Treatment.
2. The heat exchanger would be carefully designed with particular emphasis on the tube-to-header joints. Any conflict between economy and reliability would be resolved in favor of reliability.
3. There would be close inspection throughout fabrication.
4. The completed exchanger would be thoroughly checked for leaks.
5. The heat exchanger would be installed in a loop and tested at reactor conditions for an extended period.
6. After loop operation, the exchanger would again be inspected and tested for leaktightness.

A heat exchanger constructed and tested as described should have a very high reliability for many years of operation.

Water Treatment

The presence of excess hydrogen in the primary system is desirable[12-14] principally because the radiolytic oxygen content is then held at very low (often undetectable) levels by the radiation-induced water-recombination reaction

$$2H_2 + O_2 \rightarrow 2H_2O \qquad (3)$$

Owing to the sealed condition of the system, this excess hydrogen cannot be consumed by atmospheric oxygen and nitrogen (which normally would be introduced with makeup water), and it will not be lost by leakage, except possibly by diffusion through the container walls. Additional hydrogen will be produced by the over-all corrosion reactions of metals such as iron and chromium; e.g.,

$$3Fe + 4H_2O \rightleftharpoons Fe_3O_4 + 4H_2 \tag{4}$$

If it is assumed that the primary system is principally stainless steel and that the average corrosion rate is 5 $mg/dm^2 \cdot mo$,[15] it is estimated that approximately 30 standard liters (1 ft^3) of hydrogen will be produced per 100 ft^2 of surface in one year of operation. The resulting gaseous hydrogen pressure in the pressurizer (of the order of a few psi) would not be excessive.

Other primary water treatment methods used in conventional pressurized-water reactors are pH control (usually in the range 9 to 11) and side-stream water cleanup by filtration and/or mixed-bed ion exchange.[12-13,16] The need for these water-treatment procedures in the present system will be determined mainly by the degree of heat-transfer surface fouling by transportable corrosion-produce solids (crud).

pH Control. It appears that in stainless-steel and carbon-steel primary systems,[12-13,16-18] high pH (9 to 11) has a distinctly beneficial effect in reducing the quantity of corrosion products released to the coolant. Recent results[17] indicate that the release rate of corrosion products from carbon steel at a pH of 10 ($\sim$1 $mg/dm^2 \cdot mo$ at 450°F) is of the order of eight times less than the release rate at a pH of 7. Release rates for stainless steel and Inconel at a pH of 10 to 10.5 and at 500°F were $<$1 $mg/dm^2 \cdot mo$ and $\sim$4 $mg/dm^2 \cdot mo$, respectively, after 200 hr of exposure. In both cases, the results indicated that lower release rates would be found upon longer exposure. If it is assumed as the worst case that all corrosion solids released to the coolant are deposited on the fuel-element surfaces, an average release rate of 2 $mg/dm^2 \cdot mo$ in a stainless steel-Inconel primary system (pH of 10 to 10.5) in which the fuel-element surface is one-fifth the total area would result in a fouling rate of 10 $mg/dm^2 \cdot mo$. In a year of operation this would give a deposit averaging ~0.00017 in. in thickness. On the same basis, at a neutral pH, a deposit of the order of 0.001 in. in thickness (estimated from the effect of pH on the release rate from carbon steel) would result.

Thus it seems that a basic pH in the primary system is desirable, even though the over-all fouling problem is not so severe that it would be disabling in a system with moderate heat flux. The pH of the coolant could be established at the outset by the addition of lithium hydroxide or ammonia. The stability of pH with time (if ion-exchange control is absent) in such a sealed system is difficult to predict; however, since there are no base-consuming processes at once apparent, it is expected that the pH will remain sufficiently above the neutral point to be distinctly beneficial from the standpoint of fouling.

Cleanup. Side-stream cleanup by mixed-bed ion-exchange resins in conventional pressurized-water systems has three functions: (1) to remove soluble water-borne activity, (2) to filter out insoluble water-borne corrosion solids in order to reduce crud deposition, and (3) to control pH by introducing the cation-exchange resin in the lithium, ammonium, or hydrogen form. In the present system, cleanup of water-borne activity is not a primary consideration. Removal of water-borne crud probably will not be an important consideration in view of the expected low fouling rates at high pH, especially since side-stream processing may not greatly reduce fouling in any case.[19] Finally, the added pH control afforded by a mixed-bed ion exchanger will be needed only if some unforeseen base-consuming process occurs in the system. Thus it seems quite possible that a side-stream ion exchanger can be omitted.

Since the crud-removal rate could be greater, an on-stream cleanup system would offer promise of greater reduction in fouling than a side-stream cleanup system. Such a system, however, must in no way endanger reliable operation of the primary system. A means of such cleanup could be based on the fact that the principal constituents of the crud will be magnetic oxides of iron and chromium.[15] It seems possible that on-stream removal of crud, if needed, could be accomplished by using a magnetic collector in a region of low flow rate.

To summarize this discussion of water treatment, it appears likely that a hermetically sealed primary system would operate satisfactorily for a year without any treatment other than the establishment of desirable initial conditions. The preferable initial condition appears to be the use of de-ionized water which is adjusted to a high pH and which contains dissolved hydrogen. A research program involving high-pressure loops would help determine the conditions which are favorable for operation without water treatment. Operation of existing pressurized-water reactors without continuous purification might provide valuable data on crud transport and deposition. If further study indicates that side-stream purification (or possibly on-stream cleanup by magnetic collection) is desirable, its provision would have little effect on system reliability.

Secondary System

The decision to employ a conventional steam secondary system for power generation was based on the high degree to which such systems have been developed. It was recognized, however, that the conventional steam cycle will have to be simplified and improved if the ability to operate unattended for one year is to be achieved.

Three concepts of the steam system are considered in this study: (1) a "simplified conventional" steam cycle from which feedwater heaters, hot-well pump, boiler blowdown, and venting of noncondensibles are eliminated; (2) a "semiconventional" system which has the preceding simplifications plus a moisture-recovery system that avoids net loss of water from the turbine shaft seal; and (3) a hermetically sealed steam system in which water loss from the system is positively eliminated. Each of these concepts will be discussed in turn.

Simplified Conventional System

Feedwater heaters are excluded from the power system, since thermal efficiency is of secondary importance. The hot-well pump is eliminated by using a multistage boiler feed pump to perform both pumping functions. By selection of a pump which is sensitive to suction head, the level in the hot well, and, consequently, the level in the steam generator, can be controlled automatically.

Boiler blowdown can be made unnecessary for one year of operation by using a material such as Inconel for the steam generator and pretreating the water supplied to the system. With the elimination of blowdown, makeup water is needed only to replace that lost from the shaft seals. A water cleanup system does not appear to be necessary, although its provision in a side stream would not reduce the reliability of the power plant. Ejection of noncondensible gases may not be required, since the amount of gas accumulated in one year is estimated to be small enough that it could be accommodated by proper condenser design.

A brushless type of generator would eliminate the frequent maintenance required for brushes and commutators. The turbine admission valve would be the only valve required to operate after startup of the plant. The system could be of all-welded construction, with bellows-sealed valves. It would be leaktight everywhere except at the shaft seal, and water losses from the shaft seal could be accommodated by having a large water capacity in the system or by providing a makeup storage tank.

The steam pressure would be relatively low, perhaps 300 or 400 psia, and no superheating would be required. With proper turbine design the steam path would tolerate the higher moisture content of the low-pressure stages. Since efficiency is not of major importance, a relatively high condensing pressure could be used if made desirable by the nature of the heat sink and by moisture considerations in the turbine.

The equipment required for this cycle is all within the realm of present technology and can be manufactured with little or no development required. The need to replace lost water seriously handicaps this type of system, however, and precludes its recommendation for the particular application now under study.

Semiconventional Steam System

The simplified cycle described above may be further modified to eliminate the net loss of water by recovering the leakage from the turbine shaft seal and returning it to the system. Several schemes that were studied were considered to be workable. Perhaps the most practical of these is the use of a conventional bleed-off type of labyrinth seal that is vented to the main condenser, as illustrated in Fig. 5. The seal would separate the steam at the tail end of the turbine from the gas in the containment vessel. The vessel could be initially filled with inert gas at a pressure greater than that in the condenser so that a net inleakage to the system would occur. Noncondensible accumulations in the condenser would be withdrawn by a steam-jet ejector, passed over an after-condenser to reduce the moisture content, and discharged back into the containment vessel. Once equilibrium conditions were established, no net water loss from the system would result unless there were surfaces below the dew-point temperature in the containment space. If condensation were difficult to avoid, a cold surface could be provided with means for returning the condensate to the condenser through a hydrostatic leg.

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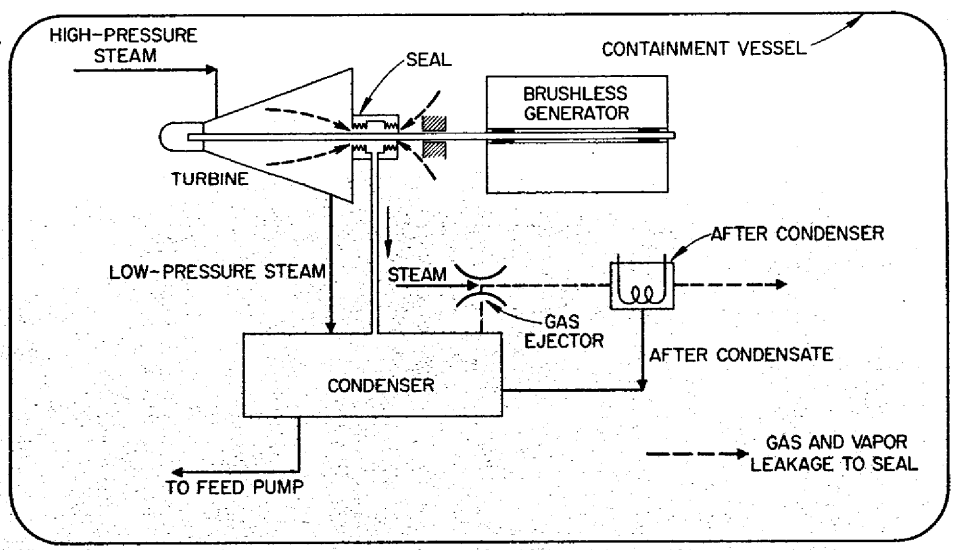

Fig. 5. Seal for Simplified Conventional Steam System.

This concept permits use of an essentially standard turbine with a brushless-generator set and, yet, largely eliminates the makeup water problem. The need for a steam ejector and after-condenser, however, adds complications to the system which are preferably avoided; thus consideration was given to a completely sealed system.

Hermetically Sealed Steam System

By using a canned turbine-generator to eliminate the shaft seals from the otherwise leaktight semiconventional plant, a hermetically sealed system which eliminates the water leakage problem can be achieved. The rotor of the generator would operate in a water or water-vapor atmosphere and would be cooled by circulation of cooling water through the stator. Generator manufacturers and manufacturers of canned-rotor pumps state that generators of this type and of the requisite size can be fabricated using presently existing technology.

Operation of the turbine-generator as a sealed unit makes it desirable to use water-lubricated bearings and to eliminate an oil-actuated hydraulic-governing system. Turbines with water-lubricated bearings have been tested by several manufacturers, and water-lubricated bearings have been quite successful in canned-rotor pumps. There has been little experience with either water-actuated hydraulic governors or water-lubricated mechanical governors, but there appears to be no major difficulty in developing such equipment. Both the governor and the admission valve could be located in the turbine housing. Other alternatives to conventional-speed governors could be based on electrical sensing of the generated frequency.

Boiler feedwater could be used for bearing lubrication and generator cooling, since it would be cool and at high pressure. As shown in Fig. 2, it could also be used for cooling the primary circulating pumps.

While the hermetically sealed system poses advanced design problems that require more development work than would be required for the two concepts previously described, the extension of present technology is not great. The hermetic seal concept promises a system of maximum simplicity with a minimum number of components. A further consideration is that, should a design and development program on this type of system meet unexpected difficulties, the effort could be diverted to the controlled-leakage concept with little delay to the over-all program. The components included in the hermetic seal concept were shown in Fig. 2.

Basic Features of the Steam Cycle

Special problems are associated with the achievement of reliability of some major components of the proposed steam system. These are discussed in this section. In addition, a discussion is included on methods of handling control functions and of eliminating the water-treatment problem.

Turbine-Generator. In order to assess the degree of reliability to be expected from conventional turbine-generators under normal conditions, Myers[3] undertook a study of the operating history of seven units of 20- to 25-Mw rated capacity. The operating history for these units, located at the Oak Ridge Gaseous Diffusion Plant, covers approximately a 14-year period. The units are conventional and are operated in a manner normal for power plants. Myers concluded that the probability of one of the units operating continuously for one year was only 0.05. In an extension of this study by the authors, the operating histories of three units of 3-Mw capacity were studied, and approximately the same order of reliability was obtained. (It should be noted that the units referred to in the above studies were not constructed to other than standard specifications. In addition, these particular units were built during World War II and consequently are not only of an older design but undoubtedly possess some compromises in materials of construction.)

A study of the maintenance work performed on the 3-Mw turbine-generator units indicated that replacement of the commutator brushes on the exciter constituted the single most frequent cause of outages. A close second in repair frequency was repacking of the turbine admission valve. In order to determine the improvement that might be realized by the elimination of certain repairs, a tabulation was prepared in which it was assumed that valve-packing maintenance and exciter-brush replacement could be eliminated. The elimination of these two repair items increased the probability of one year of successful operation from 0.05 to 0.60.

These data may be somewhat misleading, since minor adjustments or repairs are often made without shutdown of the unit; there are wide differences in opinion as to the effect of repairs made during operation. Such uncertainties in the meaning of operating data limit the value of statistical information on component performance. Nevertheless, an examination of the faults which caused forced outages of the units showed that the great majority were concerned with turbine auxiliary equipment and were not outages resulting from failures of basic components in the turbine-generator.

Although it is not uncommon for turbine-generator units to operate continuously for a year without shutdown, experience in five nuclear plants, as reported by Gilbert Associates,[1] indicates that 30% of all forced station outages are due to the turbine-generator unit. Of these outages, the majority resulted from the exercise of protective controls to prevent equipment damage. Many of the other shutdowns were concerned with repairs to the brush and commutator systems.

It appears that with the precautions of conservative design and the elimination of the brush problem, an oil-lubricated turbine-generator unit having a high probability of successful operation for one year can be developed. As discussed elsewhere in this report, however, a water-lubricated turbine-generator suitable for use with the hermetic seal concept is a preferred design and can be developed in the allotted time.

Although it is probable that the brush life experienced on conventional generator units can be extended by improved design, the predictable reliability of brushes is insufficient to permit their use in an unattended plant. The use of slip rings might prove to be satisfactory, but generators of the brushless design would be preferable, and there is sufficient technology and experience with brushless generators to warrant their consideration. Three types of brushless generators — induction, rotating-rectifier, and inductor — are discussed below:

1. <u>Induction Generator</u>. An induction motor may be operated as an induction generator by driving it above synchronous speed to obtain "negative slip." The operation of such a system on a network having a lagging power factor requires the use of capacitance in parallel with the load to obtain a leading power factor. A system of this nature has inherently poor voltage control under conditions of varying reactive load. A change in load or power factor causes a change in the excitation current, which, in turn, changes the output voltage and frequency. To correct for this shift, capacitance may be switched in and out of the system as a function of load changes. For systems with a relatively constant load, it is possible to operate with a fixed capacitance and still maintain reasonable voltage regulation. Thus load swings of 10% on a system with a unity power factor have been estimated to change the voltage less than 2% of full-load voltage. External excitation may be required during startup of an induction generator.

Induction generator units in sizes smaller than 1 Mw have been constructed and operated successfully, and there has been extensive experience with canned induction motors on primary pumps. Banks of capacitors operated in parallel with the load should be inherently long-lived and may be designed to utilize a large number of small fused units so that failure of any individual capacitor has little effect on the system capacitance.

2. <u>Rotating-Rectifier Generator</u>. A brushless generator of the synchronous type is also available. The exciter for such a machine is located on a shaft extension of the generator. The armature of the exciter is fitted with hermetically sealed rectifiers (also rotating with the shaft) to convert the a-c output to d-c current for the main generator field windings. The rotating rectifier thus eliminates the need for commutator brushes.

The silicon rectifiers used in this type of generator have good rectification efficiency, are capable of operating at relatively high temperatures, have high vibration and centrifugal ratings, and have been used extensively and successfully. They are subject to radiation damage, but it should not be difficult to protect them with shielding. Synchronous exciters with rotating rectifiers have been built in sizes up to approximately 200 kw, and, currently, at least one exciter of 1300-kw capacity is under construction.

3. Inductor Generator. Another brushless type of generator considered for this application is the inductor generator. Excitation is obtained from the stator windings. A series of staggered "teeth" on the rotor provide the path for the main pole flux, and alternating voltage is induced in the main stator windings by variation in the reluctance of the air gap. Rectifiers supply the d-c current to the excitation stator windings. Inductor generators have been built for frequencies in the range of 1 000 to 10 000 cps. The output voltage wave from an inductor generator will be influenced by a greater percentage of the higher harmonic components than is normally present in a synchronous machine, and this could affect the operation of other equipment in the system. Manufacturing experience with this type of machine, particularly for three-phase application and for lower frequencies, is somewhat limited. There is therefore some hesitance in recommending it for this application.

The operation of either an induction generator or a synchronous generator appears to be entirely feasible for one year of unattended operation. Generator efficiency for these designs is estimated to be above 80%.

An important additional consideration in the discussion of turbine-generators is the method of bearing lubrication. Conventional turbine-generator units employ oil lubrication, and the semiconventional system previously discussed does not preclude the use of an oil system. While such a system normally implies oil pumps, oil filters, and an oil cooler, the oil pump could be directly driven off the turbine shaft, and self-cleaning oversized filters could be installed in parallel to preclude flow stoppage due to clogging. As pointed out by Burns and Roe,[20] much of the complexity of a circulating-oil system might be avoided with properly designed bearings using oil reservoirs and multiple rings to eliminate the need for forced-oil lubrication. Oil slingers or deflecting vanes could be mounted on the shaft to prevent oil leakage, and cooling could be provided in the oil reservoir.

The operation of conventional turbine-generator units has demonstrated that the interchange of oil and water in the systems is essentially negligible, particularly if oil temperatures are high enough to prevent moisture condensation. Oil wiper rings effectively reduce leakage of the oil around the shaft. Water-vapor leakage to the oil system may be prevented by the use of centrifugal oil seals or other techniques. Thus oil-water vapor interchange will not prove a limiting factor for one year of operation.

An investigation was made of the possibility of using water-lubricated bearings for the turbine-generator unit. Water-lubricated bearings have been employed successfully in canned-motor pumps, and some units have had motor capacities in excess of 1000 hp. Recently, steam-turbine units with water bearings have been built and operated successfully. Units of up to approximately 750-hp capacity are currently under test and in operation.[21-22]

Water bearings use hard-surface materials, such as aluminum oxide, and thus are resistant to scoring by impurities. Bearings of both hydrostatic and hydrodynamic design have been tested. Water temperatures have been in the order of 100°F, but somewhat higher temperatures are considered to be acceptable, and widely varying water supply pressures have been employed.

In order to assess the probability that water-lubricated bearings could be successfully developed and applied to turbine-generator units within the scope of the ground rules for this study, discussions were held with manufacturers experienced with the operation of steam turbines designed with water-lubricated bearings. Discussions were also held with manufacturers of canned-motor pumps employing water-lubricated bearings. It was the unanimous opinion of these consulted that water bearings suitable for this application could be developed within the time required and that this development did not represent any major extrapolation of present technology.

Direct coupling of turbine to generator is probably preferable to the use of a geared unit, since reduction gearing would introduce an additional component that would be subject to failure. The direct coupling, however, requires a compromise between optimum turbine speed and optimum generator speed.

It is desirable that the steam flow path in the turbine be conservative in design velocities. Turbine clearances may be larger than in current practice to insure against rubbing failures, although a small loss in efficiency will result. Protective coatings can be used on the turbine blades where the moisture content of the steam is high.

Water Chemistry. One of the striking complexities of a conventional steam secondary system is the equipment required for maintaining adequate water purity in the operating system. It was felt to be of fundamental importance to the success of an unattended reactor plant that this portion of the secondary system be simplified, and therefore a study was made of the water chemistry of the secondary system. The study of this aspect of operation of the steam power cycle included consideration of the effects of corrosion products and the necessity for steam-generator blowdown, the magnitude of water losses and makeup requirements, radiolytic gas formation and venting problems, use of chemical additives, and means of cleaning up the circulating stream.

1. Water Treatment, Makeup, and Blowdown. A conventional steam plant incorporates either a continuous or periodic blowdown of the solids from the steam generator. Makeup water is supplied to replace water lost in blowdown, from vents, and in system leakage. Since the unattended reactor installation must be entirely self-contained, the need for substantial quantities of makeup water would require either a large storage facility, a water-recovery system, or a processing system for providing makeup water. The plant must be capable of one year of operation, and therefore even nominal makeup requirements would assume large proportions. Furthermore, a system for supplying or processing makeup water normally requires sensors and controls, and such components would decrease overall system reliability.

2. Corrosion. In conventional steam cycles the principal corrosive conditions arise from the presence of oxygen and carbon dioxide in the feedwater and from the leakage of these atmospheric gases into the condenser system.[23] The usual treatments are deaeration of the feedwater; the addition of oxygen scavengers, such as hydrazine or sodium sulfite; and the use of volatile basic compounds, such as ammonia, organic amines, or morpholine, to react with carbon dioxide and to increase the pH of the condensate. In the present system, oxygen and carbon dioxide initially present could be removed by purging and using suitable scavengers. If radiolytic oxygen is not produced in appreciable amounts, water control in a closed system should be considerably less difficult than in a conventional system.

The operating reliability of the secondary system is critically affected by the accumulation of corrosion-produce solids in the steam generator. In conventional systems, corrosion-product production and accumulations are minimized by maintaining a high pH in the boiler (with caustic or basic phosphate salts) and in the condenser system (with volatile amines) and often by the use of filming amines (10-18 carbon alkyl amines). Accumulated solids are periodically removed by boiler blowdown. Since blowdown is undesirable in the present system because of makeup-water requirements, consideration was given to holding solids accumulation in the steam generator to within acceptable limits by (1) the choice of suitably corrosion-resistant materials, (2) the use of permanent water additives, and (3) the use of demineralizers or other water cleanup methods.

3. Oxygen and Hydrogen Production. The production rate of radiolytic oxygen and hydrogen in the secondary system will be much lower than in the primary system, but the relative recombination rate probably will also be less because these gases will tend to accumulate in the vapor spaces. The oxygen production rate in a steam generator well shielded from the reactor flux will be due principally to the radiation from N^{16} in the primary water. For an N^{16} activity level of 100 μc/ml (which has been reported for the Idaho Test Facility),[24] the gamma energy production

rate is approximately 3×10^7 Mev/sec·ml. If this were totally absorbed by the secondary water [taking the $G(O_2)$ as 0.2],[25] approximately 6×10^{10} molecules of oxygen would be produced per second per milliliter of secondary water. If the back reaction due to recombination is ignored, this could amount to an accumulation of several milliliters of oxygen per kilogram of secondary water in a year.

This estimate is, of course, high, since not all the gamma energy will be absorbed by the secondary water, and since the recombination rate will become appreciable as the oxygen and hydrogen ejected to the steam phase are recirculated in the sealed system.[26] In particular, as excess hydrogen builds up in the secondary system from the corrosion of iron and chromium, recombination will occur more readily. Thus, with a well-shielded steam generator, no appreciable steady-state oxygen level is expected; however, it is not known what oxygen level would result if the steam generator were not shielded from the reactor.

Assuming an iron oxide—chromium oxide production rate of 10 $mg/dm^2 \cdot mo$ (which is reasonable for a stainless steel or Inconel system that includes some carbon steel),[27] it is calculated that the corresponding hydrogen production rate will be 60 standard liters (2 ft^3) per year per 100 ft^2 of hot water-steam surface area. This hydrogen will collect predominantly in the condenser vapor space. If unacceptably high hydrogen production is anticipated, a void volume can be provided in the condenser for hydrogen gas accumulation. Alternatively, excess hydrogen might be filtered off to a separate tank through a palladium metal barrier.

4. Corrosion Solids Accumulation in the Steam Generator. The order of magnitude of the corrosion-product accumulation in the steam generator can be estimated from the corrosion rate of 10 $mg/dm^2 \cdot mo$ assumed above. If there were 500 ft^2 of hot water and steam area, approximately 750 g (~1.7 lb) of corrosion products would be produced in one year. A uniform distribution of these corrosion products in a steam generator of 500-gal capacity would result in a solids concentration of 400 ppm. This is comparable with the allowable suspended-solids concentration of 250 ppm set by the American Boiler Manufacturer's Association[28] for a 300- to 450-psi

steam generator. If all solids were uniformly deposited on 200 ft^2 of heat exchanger surface, this would give a scale approximately 0.006 in. thick (400 mg/dm^2). Since not all corrosion products will be transported to the steam generator, and since they will not all be deposited on heat transfer surfaces, these numbers suggest upper limits for good materials of construction.

5. Permanent Additives. From the above rough calculation, it seems quite possible that a sealed secondary system would operate successfully for one year or longer without any water treatment at all. As in the primary system, however, it also seems likely that the quantity of transportable corrosion products could be reduced by increasing the pH of the secondary water with a permanent additive, and this should be seriously considered.

Of the various pH additives available, the most promising from the standpoint of long-term stability are (1) phosphate buffers, (2) caustic, and (3) ammonia. The first two would provide a basic pH in the steam generator. The use of caustic, however, involves some risk of caustic stress corrosion as a result of concentration by boiling. Ammonia would provide a basic pH both in the steam generator and in the condenser. The use of morpholine should also be considered. Like ammonia, it would provide a basic pH in both the steam generator and the condenser, but its long-term stability would require investigation.

In recent years, the use of long-chain alkyl amines (e.g., octadecylamine) as corrosion inhibitors in steam-plant condenser systems has met with considerable success. It has been stated that the residence time of the amine in the corrosion-inhibiting film is only a few hours,[29] and, when used in conventional plants, amines are added continuously or intermittently. At the same time, it is reported[29] that no appreciable decomposition has been found in prolonged high-temperature boiling tests in the laboratory. Thus it may be possible that the normal continuous addition of filming amines is necessary mainly because of leakage from the system. If further investigation should show that filming amines have sufficient radiation resistance and useful protective lives, they should be considered.

6. Cleanup. Since conventional boiler blowdown is not desirable in the present system, alternative cleanup methods should be considered. As pointed out above, large amounts of corrosion-product solids are not anticipated, unless, of course, significant quantities of carbon steel or other relatively corrosive alloys are used in the secondary system. A margin of safety can be provided in the design of the steam generator by making provisions for solids accumulation in low parts. Possible cleanup methods include side-steam filtration, side-stream evaporation, and side-stream or on-stream magnetic collection. Limitations of each of these methods exist. The filter may be subject to plugging (depending on the nature of the solids); the evaporator system may require automatic control; and the magnetic collector would be effective only in removing magnetic oxides. The need for recourse to these methods is not anticipated, however, unless relatively corrosive alloys are used in the secondary system.

From the foregoing it seems clear that operation of the secondary system without continuous water treatment or blowdown is quite possible. In addition, the use of permanent additives might reduce the quantity of transportable corrosion products, and side-stream cleanup methods could be employed if shown to be necessary. It is evident that investigations of the problems associated with the secondary system water chemistry should be initiated early in a program to design and construct an unattended nuclear power plant.

Steam Power Cycle Controls. Instrumentation and controls for the steam power cycle represent the major threat to reliability, according to the survey made by Gilbert Associates.[1] As stated in their report, "Failures in this category [instrumentation and controls] have produced more unscheduled losses of load than the total of all other categories." Clearly it is desirable to remove all unnecessary controls from the power system. The elimination of a reactor control system is made possible by shifting a portion of the control function from the primary system to the secondary system. Thus changes in primary-system conditions, such as temperature drifts due to reactivity changes, must be accommodated by the secondary system.

For a system in which the load is very nearly constant, the specified frequency control to within 1% is a loose requirement. On conventional turbine-generator units, where much tighter frequency control is normally exercised, hydraulic governor units act through hydraulic pilot valves to operate the turbine admission valve. Operating experience with hydraulic governor units is extensive, and such units are considered highly reliable.

If oil-lubricated bearings are employed on the turbine-generator unit, then a common oil system may be used with an oil-actuated hydraulic governor. Experience indicates that oil leakage from the governor system is negligible, and, as was the case with oil-lubricated bearings, the system can be designed to tolerate a small amount of leakage. It is imperative that hydraulic governor systems have efficient filters to remove solids from the hydraulic fluid.

Operation of an oil-actuated governing system within a hermetically sealed steam system is undesirable because of possible intermixing of the oil and water. Use of water as the hydraulic fluid with an essentially conventional hydraulic governor is an attractive possibility, since the backlog of experience with hydraulic governors demonstrates their reliability. Discussions were held with a manufacturer presently testing a water-actuated governing system on a steam-turbine unit.[22] The hydraulic system being tested is of a force-balance type similar to that for an oil-actuated system. Pressure is supplied from small pump vanes located on the turbine shaft. The degree of regulation achievable with this system is comparable to that of an oil-actuated governor, and, although operating experience is limited, it is the opinion of the manufacturer that the reliability is comparable to that of oil-actuated systems.

The water-actuated governor currently in operation contains a backup regulator that permits a second speed-regulation band. Such a design may be desirable for a reliable turbine-generator application. Two independent governor systems, one set at 1/2% frequency variation and the other at 1% frequency variation, could be employed. The wide-range governor would be inoperative if the narrow-range governor were functioning properly.

A second type of governor capable of operating within a hermetically sealed turbine-generator unit is the mechanical governor. The preference of the hydraulic governor over the mechanical governor for the majority of present-day steam turbine-generator applications is based in part on the increased precision of frequency control available with the hydraulic system. The rather loose requirement of 1% frequency variation with an essentially constant load may permit the use of the mechanical governor for this application. The essential changes required for operation of a mechanical governor include substitution of materials to permit operation in a vapor environment. A difficulty encountered in operation of turbine-generator controls is sticking of the turbine admission valve. The force required for valve operation is an important consideration in the use of mechanical governing systems where the power amplification is somewhat limited. However, this problem, which results from oxide formation on valve stems, has been associated primarily with plants operating at high steam temperatures and pressures and should not exist at the temperatures anticipated for this system.

Other methods for achieving governor control in a hermetically sealed system could be based on the electrical output of the generator. An electrical governing system usually functions by sensing the frequency output of the generator and comparing it with a desired set-point frequency to obtain an error signal; the error signal operates a position controller that regulates the turbine admission valve. A hydraulic or mechanical governor external to the hermetically sealed system could operate off a synchronous motor to regulate the turbine admission valve, and no penetrations of the sealed system would be required.

An alternate scheme for eliminating the turbine governor and admission valve was examined. This system would vary a dummy electrical load, in parallel with the normal system load, to control the turbine speed. It is envisioned that saturable reactors in series with a resistance load would vary the impedance through the dummy circuit as a function of output frequency. Solid-state components can be employed throughout the circuit. Although this system has not been developed, the advantage of completely eliminating the need for a turbine governor or turbine admission valve makes it of interest to the present application.

The exact type of frequency-governing system to be employed would require a more detailed analysis than that presented here. It appears, however, that several schemes are capable of operating successfully with a hermetically sealed system. For this application it would probably be best to undertake the development of the water-actuated hydraulic governor system as the primary effort and the development of other types of governor as the backup effort.

Even in an essentially constant-load plant, it is necessary to regulate feedwater flow to the steam generator because of short-term perturbations and longer term drifts in the system's behavior. Conventional boiler water-level controllers and feedwater regulators contain numerous sensors and control devices that would, if used, jeopardize the operation of an unattended plant.

With a constant water inventory and a fairly steady load, feedwater regulation can be obtained without the usual controllers by utilizing a boiler feed pump that is quite sensitive to suction head. The pump capacity would vary in response to the depth of water accumulated in the hot well. Performance curves for a submerged pump having such characteristics are shown in Fig. 6. This pump is designed to regulate the depth of liquid above the pump suction by the effect of suction head on the degree of cavitation occurring at the impeller inlet. Pump capacity is sharply reduced at low hot-well levels by partial vaporization of the liquid at the impeller eye. With high suction heads insuring against cavitation, the pump performance as total head vs capacity is that represented by the dashed line in Fig. 6. A reduction of suction head to below the level indicated by the dashed line will initiate cavitation. Pump capacity will then be regulated by the suction head in accordance with the solid line.

Burns and Roe[20] state that pumps of this design are currently in use in ten large-capacity utility plants. The pumps are designed to operate with cavitation and are expected to operate well in excess of a year.

The use of a pump that is sensitive to the hot-well water level in a fixed-inventory system obviates the need for auxiliary valves, level

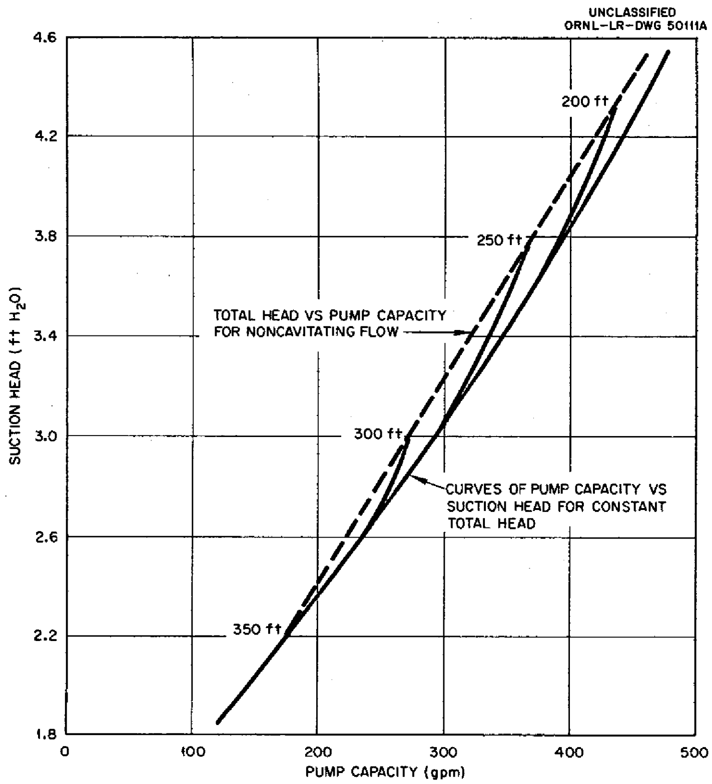

Fig. 6. Characteristic Performance Curves for a Submerged Boiler Feedwater Pump.

controls, sensors, or actuators. The importance of this to control system reliability is evident.

Other methods of achieving feedwater control and level control were examined. One scheme would employ overflow lines from a level drum in the steam generator to the main condenser or to a feedwater heater. Under normal operation the water level in the drum would remain between two overflow outlets. Thus some water could continuously recirculate to the condenser through the lower overflow, while steam would recirculate to the condenser through the upper overflow. A rise in drum level would send water through the upper overflow and increase the recirculation, while a fall in level would permit steam to exit at the lower overflow and reduce the recirculation. By proper sizing of the return lines, this system, or variations of it, could be made to keep the water level nearly constant. The above system is inefficient in that water heated in the steam generator is recirculated back to the condenser without contributing to the plant electrical output, but the loss might not be important in the present application.

Methods such as the use of eductors might also be possible for feedwater flow control. The submerged pump, however, offers greater simplicity and reliability than any other method considered.

Pumps. The conventional steam plant employs hot-well condensate pumps, boiler feed pumps, and sometimes forced-circulation pumps for the steam generator. In the plant under study, consideration was given to a system utilizing a single multistage pump to take care of all the feedwater pumping functions. The utilization of a single multistage pump capable of operating from condenser pressure to steam generator pressure has been demonstrated in the operation of SM-1. The additional modification desired for this application is the development of a reliable high-head low-flow canned-rotor pump to perform the required service. The very successful operation of the canned-rotor primary pumps in pressurized-water service indicates that the utilization of the canned-rotor pump is completely compatible with high reliability requirements. The attachment of a canned-rotor drive to a multistage impeller is a combination which pump manufacturers believe presents no difficulty or extrapolation of known technology. If the required development were not accomplished within the time available, a noncanned pump with the pump leakage controlled and bled back to the condenser could be used in a "semiconventional" system.

The possibility of incorporating the feedwater pump on the turbine shaft and thus eliminating a separate drive for the pump was considered. This may represent a satisfactory method for pumping, but the departure from conventional design might have a significant effect on the development of the turbine-generator set, and the pump would have to operate at the same speed and (if horizontal) at the same level as the turbine. Then, too, the elimination of a high-reliability pump motor probably does not give any significant increase in over-all plant reliability.

The emphasis placed on achieving utmost simplicity and reliability for the reactor directed attention to the use of an injector rather than a motor-driven feedwater pump. The relatively low temperature of condensate from the condenser (possibly 70 to 80°F) and the minor importance of pumping efficiency are conditions well suited to the use of an injector. It is particularly advantageous that injectors are of simple construction and have no moving parts. However, injectors have the disadvantages that the drop in pumping efficiency at pressures above 300 psig necessitates that large quantities of steam be used for pumping and that feedwater at temperatures above 100 to 110°F cannot be handled by a single-stage system. Present experience with injectors is such that they are not considered to be entirely reliable for unattended operation, although this opinion is in part based on the effect of load fluctuations on injector operation. Aside from questions of operability and reliability, the feedwater regulation obtained automatically with a cavitating pump makes the pump more attractive than the injector. The inherent simplicity of the injector suggests, however, that it be studied further.

Heat Exchangers. The basic heat exchanger requirement for the pressurized-water secondary system includes only the steam generator and the main condenser. The use of feedwater heaters, regenerators, reheaters, and similar equipment, which conventionally are employed to increase the plant thermal efficiency, would be largely contingent on their effect on reliability. A few heat-recovery devices would not endanger system reliability, but in general they should be kept to a minimum.

The vast experience in the construction of heat-exchange equipment indicates that units of very high integrity can be fabricated and can operate successfully without failure for periods of more than one year. As discussed in connection with the primary system, successful operation depends heavily on a high level of quality control during fabrication, but no extrapolation of current technology is required. The rather moderate conditions of water temperature and pressure envisioned for the secondary system amplify the confidence that leaktight units capable of one year of continuous operation can be constructed.

Potential materials of construction examined for heat exchanger service were the austenitic stainless steels, Inconel, and the copper-nickel alloys. Austenitic stainless steels can be used if one can insure operation of a system free of chlorides and/or maintain a very low level of oxygen. The presence, however, of even small amounts of chlorides and oxygen in such systems may result in crevice and stress-corrosion cracking.[30] On the other hand, many austenitic stainless steel systems having acceptable control of water chemistry do exist and have operated successfully for years without failure of heat exchange equipment. In this connection it should be noted that the SM-1 has operated since 1957 and has not encountered a single tube leak in any heat exchanger in the system.[11] The use of Inconel for steam generators in pressurized-water reactors has been investigated in recent work at Bettis.[27] Inconel apparently offers excellent resistance to both stress corrosion and general corrosion. It is the preponderant opinion of heat exchanger manufacturers that materials can be selected and equipment can be fabricated that can meet the one-year operational requirement.

In view of the preceding discussion, the proposed concept of a hermetically sealed secondary system with a canned-rotor pump and a turbine-generator as the only rotating machinery and a turbine-governor-frequency control as the only control system appears promising for the present application.

EFFECT OF ALTERED REQUIREMENTS ON THE DESIGN CONCEPT

The design concepts presented in this report are clearly a sensitive function of the requirements for the nuclear power plant. The purpose of this section is to point up those design areas that would require careful re-examination to meet changes in the initial specifications. Of particular interest are the effects of the following assumptions: communication with the plant is possible; size and weight are considerations; unattended operation beyond one year is important; the plant is not expendable; the time schedule is different; the load is variable.

Communication with the Plant

If communication with the plant can be achieved, information pertinent to plant operation can be monitored. System temperatures, electrical frequency, and perhaps flow information could be transmitted to a receiving center where elementary corrective actions could be initiated by simple signals. Fixed-increment adjustments of control rods would permit correction for long-term reactivity changes indicated by changes in water temperature. Similarly, fixed-increment adjustments of a governor system could correct for drift of the set point. Communication also would permit nonoperating spares to be remotely energized or emergency shutdown of the plant to be initiated when received information indicated the need. The hazard of misoperation and failure because of malfunctioning of the communication system would, of course, exist, but precautions, such as the use of coincident signals, might be employed to prevent spurious actions. Thus provision of a system for communicating with the plant might permit improvement of plant reliability.

Much of the broad simplification proposed to achieve reliability for the unattended plant would be useful in designing a plant for operation with small crews. Some of the modifications proposed for the control and water-treatment systems might greatly reduce the operation and maintenance requirements of an attended plant.

Plant Size and Weight

Restriction of the size and weight probably would alter the design of a reactor power plant in a manner that would adversely affect its reliability. The design considerations imposed by moderate restrictions might have little effect, whereas overriding considerations of size and weight would likely render reactors such as the pressurized-water concept unsuitable. In designing a pressurized-water reactor for limited size and weight, increased system pressure, temperatures, and flow velocities, higher reactor power density, and greater turbine-generator speeds would have to be investigated. Although incorporation of many of these changes would not necessarily decrease system reliability, they might remove the design from the realm where experience with present pressurized-water reactors is applicable.

Long-Term Reliability and Expendability

The design of an unattended power plant capable of operating several years on a continuous basis imposes even more severe restrictions on the systems and components. Particularly sensitive to longer term operation would be the water-treatment requirements. Simple systems to permit introduction of chemical additives might be needed, and side-stream purification would probably be necessary.

The problem of building in and controlling sufficient reactivity for longer term operation might necessitate the use of an automatic control system, although it is possible that the lifetime obtainable using burnable poisons would extend well beyond one year. If automatic control were required, it might be that a programmed shim-control system operated by a clock or a watt-hour meter (as mentioned under Reactor Control) would be more reliable than a system which employed feedback from the reactor. Almost certainly, longer term reliability would be achieved if one proceeded to build plants initially capable of one year of operation and then modified them as operating experience dictated.

Two major changes in concept would be required if the plant, or even its major components, were not considered to be expendable. Equipment such as the turbine-generator and the pumps would require protective devices capable of shutting down the plant, and deposition of activity might have to be controlled so that core components would be repairable. These changes, particularly the use of protective devices, would decrease the probability of the plant running a year without shutdown.

A compromise between expendability and repairability might be attractive for some applications. This could involve construction of the plant in packages (say a core package, turbine-generator package, etc.), which could be replaced as units. The cost might be much less than that of replacing an entire plant.

Variable Loads

The effect of requiring an unattended plant to operate successfully on a variable load might significantly change recommended components and control concepts. The use of an induction generator, for instance, with a variable load might produce voltage variations beyond the desired limits. Similarly, certain frequency control systems (such as the mechanical governor) might be less adaptable to a variable-load plant. The effect of sudden load changes on reactor control requirements would have to be examined.

If the load were intermittent in nature, the requirements for the plant would be even more stringent. Both the primary and the secondary systems would have to be capable of going to power unattended. Provision of a variable dummy load to preclude zero-power operation and to eliminate severe load changes, if necessary, might improve the reliability of a load-following system. Small or gradual load changes not requiring plant shutdown could probably be accommodated by the design concepts initially proposed.

Effects of Time Schedule

Certainly one of the most fundamental restrictions affecting the design concept is the time within which the plant is required. This is emphasized by the suggested effects of time changes on the concepts presented. If production of an acceptable system in three years were required, it is probable that success of the program would not be based on design modifications beyond the semiconventional system proposed in this report (although the hermetically sealed secondary system would be attempted as a secondary effort). If, on the other hand, four to five years were allowed for procurement of an acceptable plant, the hermetically sealed plant would certainly be utilized. For periods beyond five years, one could afford to deviate to still greater degrees from known technologies and proven concepts.

SUMMARY AND CONCLUSIONS

Power reactors are normally designed for constant attention and continuous maintenance, and there are no existing reactors capable of producing electricity for one year unattended. Not only are complete power plants incapable of one year of operation without maintenance, but few of the individual components have the requisite reliability. This is particularly true of control systems and of steam power systems, for which regular maintenance is accepted practice. It is less true for the components in the primary system of pressurized-water reactors, because the Navy program has led to the development of reliable components.

Since existing systems do not have the required reliability, the feasibility of achieving an unattended reactor in four years actually involves the question of whether a reliable system can be developed in the allotted time. The four-year period is so short that there would be little opportunity for perfection of new concepts, and the authors believe a program to achieve the stated objectives should, wherever possible, be based on proven concepts and proven technology. System reliability is most likely to be achieved by simplification to the point that operation depends on a minimum number of components. The objectives of this study make simplification particularly promising, since the system is not required to be repairable, have a long life, follow a varying load, or operate in a populous area. Although reliability is not consistent with minimum cost, elimination of equipment by simplification may offset the increased cost of individual components.

A pressurized-water reactor fueled with highly-enriched uranium was selected for this application because of the successful experience with it and because it is amenable to extreme simplification. The selection of the pressurized-water concept is not intended to suggest that this is the only or even the best system for this service. It is, however, the one which appears most likely of achievement in a limited time.

A reactor plant based on the concepts which evolved from this study would consist of relatively few components. The only mechanically operating components would be the primary coolant pump, the boiler feed pump, and the turbine-generator set; the turbine governor would be the only control system. This simplification is obtained by making the following changes in an ordinary pressurized-water reactor plant:

1. The reactor is designed to operate after startup without a nuclear control system. Burnable poison is used to limit gross reactivity

changes, and the negative temperature coefficient is used to compensate for changes which are not eliminated. The effect of allowing the core temperature to vary (say over a range of 50°F) is to shift the reactor control problem to the turbine governor. Since the governor is already required for frequency regulation, no additional components are made necessary.

The design of a turbine governor and admission valve is eased by elimination of the normal requirement of tight frequency control over a wide range of loads. Although several existing types of governor could be modified for this system, the reliability of a governor is one of the major system uncertainties. (The use of a variable dummy electric load to control frequency without a throttle valve or a governor appears to merit serious investigation.)

2. An automatic control system for the pressurizer is avoided by a design which permits the electric heaters to remain on continuously.

3. Boiler feedwater controls are eliminated by use of an existing type of cavitating pump that is sensitive to suction head. The pump will maintain a constant water level in the hot well and, consequently, in a fixed-inventory system, will maintain a constant level in the steam generator.

4. Both the primary and secondary systems are hermetically sealed, and there is neither water treatment nor blowdown. It appears feasible to operate for a year in this manner without encountering disabling problems from corrosion or crud buildup.

The operation of a hermetically sealed secondary system requires the use of water-lubricated turbine-generator bearings and a generator which will operate in a vapor or water environment. Several proven brushless generators are capable of such operation. While only a few turbines in the size range needed have been built with water-lubricated bearings, canned-rotor pumps employing water bearings have been successfully operated for many years. (If development of water-lubricated bearings does not proceed rapidly enough, an oil-lubricated turbine with a brushless generator can be accommodated in a modified design.)

The remaining components of the power plant appear to present no reliability problems. Fuel elements with lifetimes longer than required are available. Primary circulating pumps have a demonstrated history of reliability. With careful design, selection of materials, fabrication, and testing, the piping, heat exchangers, and pressurizer can be made to be reliable.

It is probable that using the concepts proposed, reactors capable of unattended operation can be available in four years. To achieve this objective, a conceptual design study must progress rapidly to the beginning of detailed design, in order that the construction of a prototype can be accomplished early in the program. Studies of water chemistry, core physics, and reactor control, and the development of a frequency-control system, turbine-generator, and boiler feed pump should begin immediately. A well-integrated and rapidly moving program can lead to useful power plants in the allotted time. Experience with the first generation of reactors and continuing development will increase the reliability beyond what may be achieved initially.

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