STEAM COOLED POWER REACTOR EVALUATION—BELOYARSK (URAL) REACTOR

April 1961

Hanford Works
General Electric Company
Richland, Washington
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STEAM COOLED POWER REACTOR EVALUATION -

BELYARSK (URAL) REACTOR

For

Evaluation and Planning Branch
Division of Reactor Development - AEC

APRIL, 1961

Hanford Works

NPR PROJECT
IRRADIATION PROCESSING DEPARTMENT
GENERAL ELECTRIC COMPANY
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BELOYARSK (URAL) REACTOR

I. INTRODUCTION

The primary purpose of this study of the Beloyarsk Station design is to evaluate the effects of nuclear steam superheating, as utilized in the Russian design, on the economics of electric power production, assuming the use of such a design in the United States.

A reference boiling-water reactor utilizing the features of the Russian design, that is graphite-moderated and with internally cooled fuel elements, is non-existent. The magnitude of the benefits of superheating as demonstrated in this design can, therefore, be derived only analytically since the scope of the study does not include the complete development of a reference non-superheating design. The analytical work on this subject, performed as a part of the study, is not comprehensive but serves to indicate the order-of-magnitude of the benefits of superheating if it is assumed that the design is near optimum for boiling as well as for boiling and superheating.

In the appraisal of the Beloyarsk plant, we have accepted the claims made in the Russian references where they cannot be checked and where they appear to be reasonable, checked data where possible, and commented in those cases where the claims seem to be unreasonable or where we feel that the design is particularly risky.

The Russian design has been modified in only four respects, in order to make it conform with the study ground rules:
1. The thermal rating has been tripled to reach 300 MWe net by doubling the number of fuel assemblies and lengthening them, and by increasing the coolant tube diameters and fuel cylinder diameters slightly to maintain the original coolant pressure drop through the element with a flow 1.5 times that of the original Russian design.

2. The reactor and boiling loop components have been placed inside a vapor suppression sphere.

3. The control rods have been adjusted in number and location to permit fuel exposure levels of up to 11,000 MWD/T, and

4. In the primary case, the wall thickness of the stainless steel tubes in the fuel assemblies has been increased (about doubled) to conform with American code requirements.

The design of the enlarged reactor retains the distinctive Russian characteristics. In the future (1975) plant, the improvements which were considered were those associated with cost savings in manufacturing the components of the Russian plant, with applying the reactor materials and fuels less conservatively but under conditions judged to conform with American practice at that time, or with design changes not affecting the fundamental character of the reactor. Radical changes in the Russian design for purposes of improving it have been avoided, again with the purpose of retaining fundamental integrity of the original design.

The scope of this particular study did not include any experimental, test or demonstration work, on either specific components of the Russian design or on superheating technology in general. Our appraisal of the problem of nuclear superheating, as it is carried out in the Russian plant, is made against the general background of American steam superheating and nuclear knowledge and experience.
A limitation on the value of the appraisal is that part of the information on which it is based is some two years old, and may in fact, already partly be outdated. This information, however, is all that was available. Late in the course of this study, a translation of a draft of a paper given at the 1960 Vienna Conference became available. That document emphasized the merits of a 9 percent molybdenum uranium alloy fuel material above those of others being tested in APS-1. It can be inferred that this is the element being used in the Beloyarsk reactor. Our report covers both the molybdenum alloy and the magnesium matrix fuel described in the earlier references. As a part of this study, cost estimates were presented to the AEC for:

1. Molybdenum alloy and magnesium matrix fuels.
2. Fuel exposure levels of 2200, 5500, and 11,000 MWD/T.
3. Stainless steel pressure tube wall thicknesses in fuel elements as in the Russian design, and as increased to meet American code requirements.

The primary evaluation, used in the comparative study by the Commission and incorporated in reference document,* was for magnesium matrix fuel, at 11,000 MWD/T, and with American code wall thickness in the fuel element pressure tubes.

It should be understood that this report is concerned only with the one design, and does not explore the general merits of graphite-moderated, water-or-steam-cooled reactors for electric power production.

II. SUMMARY

BRIEF DESCRIPTION

The reactor of the Beloyarsk Station is a graphite-moderated, uranium fueled reactor having vertically oriented fuel channels. The fuel is internally cooled in pressurized tubes by boiling (light) water in approximately 75 percent of the total channels and by steam in the remaining 25 percent, the fuel elements in both groups being of identical design.

The core, as it has been modified for this study, consists of a graphite structure having 1980 fuel assemblies in a 7.874 inch square lattice. The graphite is surrounded by a graphite reflector, a tank made of low-alloy steel plate for nitrogen atmosphere confinement, and a biological shield consisting of a compartmented water tank.

A distinctive feature of the reactor is its fuel element, a bayonet assembly inserted from the top of the reactor. The assemblies carry, in addition to the uranium fuel, all of the internal reactor piping and a part of the graphite moderator. Cooling water is fed into the fuel element head through the pressure piping passing over the top of the reactor. From the head inlet, it passes to a central stainless steel tube in which it is carried to a distribution chamber at the bottom of the fuel element assembly. From this chamber the primary coolant flows upward through six small stainless steel tubes surrounded by fuel material consisting of finely divided metallic uranium in a magnesium matrix, or consisting of 9 weight percent molybdenum uranium alloy. The fuel and cooling tubes are embedded in a graphite cylinder 3.23 inches in diameter and 23.6 feet long. The fuel is cooled internally by boiling water in 1448 tubes. The primary coolant at 36.4% bulk steam quality leaves the reactor through the fuel assembly head and passes
to a steam separator. Steam from the separator passes to a heat exchanger in which water in a separate loop is boiled. This secondary steam is fed to the 532 superheating fuel elements. The steam cooled fuel elements are of the same design as the boiling water fuel elements and operate at the same general power levels. Steam leaves the superheating fuel elements and is piped to the turbine. At the turbine throttle, the steam has a temperature of 932 F at a pressure of 1280 psia.

Water from the steam separator and condensed steam from the primary side of the heat exchangers pass to a preheater where heat is transferred to the feedwater. The feedwater leaves the second section of the preheaters as 27.4 percent quality steam and flows to the evaporator where the steam and water are separated. The steam flows to the superheater and the water serves as feedwater to the evaporator.

The reactor control rods enter from the bottom of the core. The control system consists of 162 manually operated rods used to compensate for burnup in the reactor and provide shutdown control. There are in addition 55 compensating and fast scram rods and 24 rods for regulation of power level of the reactor. All control rods are water cooled.

The reactor is equipped with the necessary piping and controls to start up with the superheating channels water cooled until the reactor has reached approximately 30 percent of rated power. At that level, a transition takes place in which steam from the evaporators expels the water from the superheating channels.

All parts of the primary or boiling coolant recirculating system are of stainless steel. The piping in the superheating parts of the system conforms
with ordinary power plant practice - the only stainless steel in this system being that in the fuel elements themselves and the coolant supply and return connectors.

The reactor and all parts of the boiling water loop are enclosed in a confinement sphere which is used in conjunction with a vapor suppression system to control fission products in event of a nuclear incident. Water in the fuel discharge reservoir is used for the vapor suppression heat sink to cool gases and vapors expelled from the reactor, and from the equipment cells located on each side of the reactor.

The more complete description used for purposes of preparing the capital and operating cost estimates is given in Section IV and in the drawings.

ASSUMPTIONS MADE FOR THE PURPOSES OF THIS EVALUATION

It has been necessary to make a number of assumptions regarding the design and operation of the plant for purposes of this evaluation. Chief among these are:

1. The fuel material used in the reactor is the finely divided uranium in a magnesium matrix (described in References 1 and 2) or a 9 percent molybdenum uranium alloy.

2. The finely divided uranium is in the form of uranium shot.

3. The stainless steel forgings at the top end of the fuel element assembly can be reused four times.

4. Disassembly of the fuel elements and removal of graphite is performed at the reactor plant.

5. Individual fuel element flow monitoring is not required.
6. Poison injection in the control rod thimbles will serve as a backup reactor control.

7. Stainless steel in the fuel elements will be type 321 (or the nearest American equivalent to Russian type 1X18H9T), and in the loop piping 304 or 316 whichever is more economical.

8. The control rods will be of boron stainless steel.

9. Diesel pumping power will be provided for emergency cooling.

10. At 2200 MWD/T (metric tons are used throughout the report), all of the fuel in the reactor will be replaced approximately every 2 years, and at 5500 and 11,000 MWD/T, 25 and 12.5 percent, respectively, of the fuel in the core will be replaced each year.

Miscellaneous assumptions, such as the necessity for containment, will be made to comply with the ground rules.

EXCEPTIONS TO GROUND RULES
The study has been carried out under the ground rules enumerated in the letter to K. A. Dunbar from F. K. Pittman, "Steam Cooled Reactor Studies," with the following exceptions:

1. Standard fuel fabrication costs are not used for the fuel element since the fuel element is entirely different from those for which standards have been established. The fabrication costs have been estimated by HAPO Fuels Preparation Department, the estimates being based on an assumed fabrication process described in the report.
2. A full-flow demineralizer is not used in the boiling loop of this reactor; bypass demineralization is used since the working fluid is recirculated in a closed loop without going through the turbine.

COST ESTIMATES

In the performance of this study, the capital operating and fuel costs were estimated and reflected in document KE-60-31, "Study of Steam-Cooled Nuclear Power Plants Capital and Power Generation Costs," for the AEC by Kaiser Engineers, December 31, 1960. The fuel conditions used for purposes of that report are:

- Magnesium matrix type
- 11,000 MWD/T
- Code pressure tube wall thickness
- Pressure tube material, 304 stainless steel.

A later study of the alternative Mo-ally fuel indicated that the use of that type of fuel would increase fuel costs by about 7 percent, and increase the unit cost of electric energy by about 2 percent.

A later study of the use of type 321 stainless steel instead of 314 indicated a cost increase of about 4.5 percent in the fabrication cost of fuel, and about 0.4 percent in the unit cost of energy.

PRINCIPAL CONCLUSIONS

The principal conclusions reached in this evaluation of the Beloyarsk reactor design are as follows:

1. An examination of the design does not indicate any areas in which the Russian technology is more advanced than that of the USA. One exception may be in the rather unique application of stainless steel
in the fuel tubes. The steel mentioned in the references, type 1X18H9T, is an 18% Cr., 9% Ni, 1% Ti alloy similar to American AISI 321. This material is used in very thin-walled pressure tubes in the fuel elements. The risks of operating at the high stresses computed for such tubes, and the careful manufacturing, testing and operating controls necessary to prevent failure, would be such as to influence American designers to use a wall thickness about double that of the Russian design, meeting code requirements and lending more assurance of successful performance. Metallurgists doubt that the Russian steel can be superior enough to warrant the risks entailed in use of the thin-walled tubes.

2. The Russians apparently lag the Americans by a few years in the enthusiasm with which they view the use of fuel materials such as the magnesium matrix and the 9 percent molybdenum uranium alloy. The enthusiasm in this country for such fuels has abated considerably in recent years. Internal cooling of fuel elements in a graphite moderated reactor as utilized in this design, does, however, place a premium on good heat conductivity in the fuel material and there is an incentive to develop such fuels rather than turning to ceramic types. The use of a ceramic or low conductivity fuel would require the addition of some separate cooling system to prevent excessive graphite temperatures.

3. Except for the stainless steel in the fuel tubes, reactor materials are applied conservatively. Aside from the question of the rate of coolant fuel-material reaction in the event of a process tube failure, and possible bad effects on the coolant stream and inside the core, the prospects of successful operation of the reactor appear to be good.
The Russian references state that fuel failures have not occurred in operation, and that the consequences of failure under test were not serious. It is felt that these data would have to be corroborated here before reliance could be placed on such a design.

4. Bottlenecks to achieving high specific power exist by reason of internal fuel cooling as used in this design, but there are inherent compensating advantages in reactor physics.

5. With the fuel exposures noted in the 1958 Geneva paper for magnesium matrix fuel, 2200 MWD/T, the energy costs as computed in the USA would be prohibitive. At higher exposures (5500 to 11,000 MWD/T), which we feel could be achieved in this reactor and which are reported in a Russian 1960 Vienna Conference paper, the estimated energy costs equal those of the lower and medium cost types in the 1959 AEC study on "Current Status" of non-superheating nuclear plants.

6. Some of the problems of nuclear superheating such as flooding coefficients, changes in power distribution with burnout, and deposit of contamination on fuel elements, have been avoided or rendered less difficult by the design of the Russian plant (in some respects because of the characteristics of graphite as a moderator) which augment the prospects of successful operation of the plant. Although these features of the design have increased capital cost to some degree, they should facilitate early achievement of nuclear superheat on a reasonably economic basis.

7. The use of nuclear superheat appears from rough analysis to have cut about 1 mill/kw-hr from the energy costs of a hypothetical boiling water reactor having the same general characteristics and operating fuel in the 5500 to 11,000 MWD/T range.
8. The potential of a reactor utilizing internally cooled fuel appears to be moderately good. Problems exist with the cooling of graphite and the outer zone of the hollow fuel cylinders, but the design is promising economically because of good physics and adaptability to economic steam conditions. In our opinion, some further work should be carried out with internal cooling, or with internal and external cooling, making use of the presumably more advanced American fuel-material and graphite technology, investigating the use of materials such as Inconel for fuel tubes, and seeking a balance of parameters closer to the optimum in our economy.

9. The general conclusions of the original study are not changed appreciably by substituting molybdenum alloy for magnesium matrix in the fuel element. Between the two fuel materials there are, however, some differences as to the prospects of successful and economical operation.

(a) The reaction rate between the alloy and water (or a steam-water mixture) at the specified operating temperatures, for unirradiated material, is much lower than between magnesium matrix and water. If this holds true for irradiated material, the consequences of a fuel tube failure should be least serious with the alloy fuel material.

(b) Fuel swelling with the alloy might possibly be more of a problem than with the matrix, particularly in the superheating tubes in which the maximum fuel operating temperature is highest. In this regard, the prospects of reaching high fuel exposures may be less than for the matrix type.
(c) For equal exposures (11,000 MWD/T), the fuel costs of the alloy type element are about 5 percent higher than for the matrix because of (1) the higher enrichment required, and (2) the consumption of molybdenum. Apparently the Russians feel that alloy has good prospects for operation beyond 11,000 MWD/T and expect that type of fuel to pay out in long exposure.

**DISTINCTIVE CHARACTERISTICS OF THE BELOYARSK PLANT AS THEY AFFECT ITS ECONOMICS**

The distinctive characteristics of the plant are:

1. Fuel consisting of finely divided metallic uranium in a magnesium matrix, or 9 percent molybdenum-uranium alloy.

2. Internal cooling in thin pressurized stainless steel tubes which are an integral part of the fuel element.

3. Identical design for the boiling and superheating fuel elements.

4. Superheating in intermediate zones of the reactor core.

5. Separation of the boiling and steam superheating coolant loops.

The following comments apply to the reactor in general or to the reactor with magnesium matrix fuel, except where it is stated that they apply to molybdenum alloy fuel.

As discussed in more detail in Sections V-A and V-D of the report, the fuel element has physics characteristics which are desirable from an economic standpoint in that the requirements for initial enrichment are low, and the conversion ratio reasonably high, tending to keep burnout costs and inventory costs low and the quantities of residual saleable Pu high relative to other
boiling superheating reactors. On the other hand, the conditions for heat removal are less desirable in that the reactor, at its rated power level would operate at low specific power thus tending to increase fuel inventory costs, and to increase the capital cost of the reactor because the core has to be large for its power rating.

The use of internal cooling, resulting in small-diameter pressure tubes, permits the use of high steam pressures without the necessity for large amounts of neutron absorbing materials in the core. This reflects not only in the desirable nuclear performance indicated above (with consequential good effects on fuel costs) but also in a moderately high thermal efficiency (35.4% over-all). Both of the above effects are favorable from a cost standpoint.

On the other hand, the fact that the stainless steel tubing is wasted with each fuel replacement adds to fuel cost, and becomes a large cost item at low fuel exposure levels such as the 2200 MWD/T level given in the earlier references as an assumed operating level. This element of cost is not too detrimental at exposure levels in the 5500-11,000 MWD/T range. On the whole, costs for these fuel elements were found to be somewhat less than had originally been anticipated. The assumption that the stainless steel top end fitting could be reused has acted to reduce fuel costs, but unit fuel costs at 11,000 MWD/T would not have been prohibitive even if the fittings had been discarded after the first usage.

Internal cooling as applied in this design is economically disadvantageous in that it imposes limits on specific power. The graphite, in which some 5 percent of the heat is released, has to be cooled by heat transfer to the fuel, thence to the coolant. High temperatures are encountered in both the
graphite and the outer zones of the fuel cylinder, which tend to limit increases in specific power. In judging the magnitude of the various good and bad effects, one has to consider whether the good physics and adaptability for favorable steam conditions are more beneficial economically than the low specific power would be detrimental.

Regarding point (3), identical design for boiling and superheating fuel elements, this characteristic adds flexibility to the reactor, and perhaps in a small way beneficially affects fuel costs through requiring only one fabrication process and avoiding the necessity for segregation of two types of fuel throughout the cycle.

Item (4) in the above listing, superheating in intermediate zones of the core, has no decisive effect on economics. If the power of the reactor were to be increased, it is possible that an economic bottleneck in the cooling of the superheating elements in this portion of the core might be reached sooner than if they were in fringe zones. (See discussion on heat transfer.) Locating the superheating elements in a zone least sensitive to the presence of water in the cooling channels helps the safety and control systems, where it contributes some inherent stability which tends to keep to a minimum the capital expenditures otherwise necessary to make the reactor safely operable.

Item (5), separation of the boiling and steam superheating loops, adds to capital cost. A heat exchanger boiler, in which steam in one loop produces steam in another separate loop is provided, which would be unnecessary if steam from the boiling water elements were routed directly to the superheating elements. The practices of the Russians in this regard avoids some of the problems of depositing contaminants from the boiling water loop on
superheating fuel surfaces which is desirable from an operating standpoint, but does add to capital cost.

Generally speaking although the Beloyarsk reactor has many good qualities, its economics at the originally rated operating levels, both with regard to the power and fuel exposures, would be poor in the USA. At higher fuel exposure levels, the economics would be comparable to the best of the USA 1959 "Current Status" non-superheating plants. If economics are ignored, the prospects for successful operation would, with the exception noted, be good in that the materials of the reactor and fuel are being used conservatively. It appears that there would be good prospects for somewhat higher specific powers as discussed under Future Potential, and for fuel exposure levels up to 15,000 MWD.

Plutonium Recycle
An effort was made to determine how the Beloyarsk reactor design compared with some American power reactors regarding characteristics for economic use of plutonium as a fuel. The study indicates that this reactor may utilize Pu fuel economically at about $1.4/gram fissile nitrate; about the same as was found in a similar analysis for a water-moderated reactor (APWR).

Resume' of the Problems and Advantages of Nuclear Superheating as Exemplified in the Russian Reactor
The problem of steam-water separation is simpler in this reactor than in some other concepts due to the external separation and isolation of the boiling loop.

The problem of designing superheating fuel elements that can operate at the high temperature necessary for this service exists with this reactor as much
as any other. The magnesium matrix is not particularly well adapted to this
service at higher specific powers. As has been remarked elsewhere, the
melting point of the magnesium matrix is fairly low and would in all
probability be a bottleneck in trying to achieve a relatively high specific
power with this design.

The problem of insulating the process tube from the moderator does not exist
in a graphite moderated reactor. There is the problem, however, of cooling
the graphite moderator when one attempts to get high specific powers as
discussed previously. The problem of emergency shutdown cooling of the
superheating fuel elements exists in this reactor as in other superheating
types. It would seem, however, that this problem can be solved rather
easily considering the provisions in the design for natural convection of
water in the boiling loop to generate steam for cooling of the superheat
chambers, and final dissipation of the heat via the emergency condenser.

Power shift with burnup does not appear to be a serious problem in the
design of this reactor. As discussed in the Physics Section, it should be
possible to maintain the relative reactivity of the zones of the reactor
such that the magnitude of the power shift will not be objectionable. The
flexibility which is gained by the large number of control and compensating
rods will have a part in maintaining the proper balance. It should be
mentioned also that the method proposed for controlling the flow of the
superheating steam is such that it can accommodate moderate power shifts.

Economic Benefits of Superheating

A brief study was made to estimate the worth of superheating for a nuclear
plant of this type. The economic benefits of superheating were found to
be as follows:
<table>
<thead>
<tr>
<th>Fuel Exposure Level, MWD/T</th>
<th>Differences Between Unit Energy Cost in Superheating &amp; Saturated Steam Plants - (mills/kw-hr)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2200</td>
<td>1.4</td>
</tr>
<tr>
<td>5500</td>
<td>1.0</td>
</tr>
<tr>
<td>11,000</td>
<td>0.9</td>
</tr>
</tbody>
</table>

It is again emphasized that this analysis is rough, and is made with the assumption that such a boiling water plant would be near optimum.

The differences shown are the result of an 11.5 percent increase in capital cost and an 11 percent higher fuel cost. It was assumed that operating and maintenance costs would not change.

The steam flow in the saturated plant will be about 50 percent greater than in the superheat plant due to a poorer turbine cycle efficiency and a lower turbine throttle enthalpy. This will require an increase of about the same amount in reactor primary coolant flow. About 75 percent of the increased capital cost is in the reactor plant account and is the result of providing equipment to handle the larger flow. The remaining 25 percent of the increase is in the turbine-generator account and is primarily due to providing a turbine to operate with saturated steam.

The major part of the increase in fuel cost results from the lower turbine cycle efficiency with saturated steam. Auxiliary power requirements are also about 1 percent higher due to increased pumping power in both the reactor and the turbine systems.

**Research and Development - 1967 Plant**

The estimate given above for the 1967 plant is based upon a design concept which utilizes only "current technology". The interpretation of what
current technology is, relative to a magnesium matrix type fuel element, may be subject to question. Most of the work on the magnesium matrix element performed here at Hanford took place some 5 years ago when the interest in this type of fuel was quite active. Later on the interest in this type of fuel fell off. As discussed in the report, there have been a number of tests on these elements. If the use of such fuel material were to be again seriously considered, it would be necessary to reinstate a rather complete research and development program on this subject. Referring to the molybdenum alloy element, it would also be necessary to extend the experimental work performed by others so far on this type of fuel material.

There is a particular need to test the fuel element at the temperature and pressures at which it will actually operate to prove the performance of the thin-wall stainless steel tubes and the conditions of heat transfer between the tubes and the fuel material. It will also be necessary to make simulated tests of failures in the walls of the pressure tubes to determine whether or not the products of fuel corrosion will back up into the coolant stream, and to determine the mechanism whereby the pressure inside the fuel material will be relieved in case of either a major break in the pressure tube wall or a pinhole leading gradually to a more complete failure.

Some exploratory research will be required on corrosion and radioactivity carry-over.

From our studies, it would appear that the work done by the Russians on flow stability was thorough and the testing work in this connection probably need be only a general check without going into a comprehensive testing program. It would be necessary to conduct confirmatory work on core physics, and equipment development work on some instrumentation and pressure fittings used on the reactor.
The research and development necessary to support design and construction of the 1967 plant, the main portion of which is enumerated above, is $5 million dollars.

**Future Potential (1975 Plant)**

The future potential of this reactor, still adhering to the basic Russian design and making only those types of changes discussed in the Introduction, lies in the following areas:

1. Increase the specific power, i.e., make the necessary changes to operate the core described above at power levels up to about 150 percent of the reference thermal level. Obtain this increase by increasing the coolant flow rate and operating at somewhat higher temperature limits in the reactor core, approaching the limiting condition for temperature in the outer zones of the fuel element, considering swirling heat transfer in the boiling elements if transfer from the tube surfaces begins to be a bottleneck.

2. Operate the fuel at 15,000 MWD/T adding such control rods as are necessary to achieve operation at this fuel exposure level. This may not be an optimum fuel exposure but would represent an improvement in the operating economics of the reactor.

3. Delete the evaporator and preheater and make necessary changes in the superheat steam control and feedwater systems, thus decreasing capital cost.

4. Consider the use of carbon steel rather than stainless in the exterior piping for the boiling water loop.

5. Replace the canned rotor pumps in the boiling water loop with pressure sealed pumps.
6. Consider one reheating cycle, carrying out the reheating in tubes in the same zones as for superheating.

Note: The use of vibratory compacted oxide in the fuel space instead of magnesium matrix or molybdenum alloy could also be a subject for study, but would require reactor changes too extensive to be even roughly estimated in the scope of this particular study.

The net effect of each of these potential improvements on the economic performance of the reactor is estimated to be a reduction of approximately 20 percent in the unit energy costs given herein for the 5500-11,000 MWD/T fuel exposure cases.

The research and development effort necessary to support such a program would follow the same general lines indicated for the 1967 plant and is roughly estimated at $3 million addition to the R&D costs for the 1967 plant.

The potential improvements are discussed in Section VII.
III. SUMMARY OF REACTOR PLANT CHARACTERISTICS

Reactor Type: Russian Superheat

DESCRIPTION

A. Heat Balance$^1$

1. Net Plant Power, MWe 300
2. Gross Turbine Power, MWe 320
3. Total Reactor, MWe 847.6
4. Net Plant Efficiency, % 35.4

B. Turbine Cycle Conditions$^1$

1. Throttle Temperature, °F 932
2. Throttle Pressure, psia 1,280
3. Total Steam Flow, lb/hr $2.622 \times 10^6$
4. Condenser Back-pressure, in Hg Abs 1.5
5. Final Feedwater Temperature, °F 419
6. Number of Feedwater Heating Stages 6

C. Reactor Description

1. Reactor Vessel
   a. Inside Diameter, ft. 39.36
   b. Inside height, ft. 36.70
   c. Wall Thickness, in. (cylindrical portion) 1
   d. Material SA 301 B
   e. Design - Pressure, psig 5
   f. Design - Temperature, °F 1,100

$^1$ The data for the steam utilization and electric power generation portions of the plant were developed independently of those in the Kaiser Report, KE 60-31, "Study of Steam-Cooled Nuclear Power Plants Capital and Power Generation Costs," 12/31/60. They differ slightly from those in the reference report but are generally confirmatory.
2. Reactor Core

a. Active Equivalent Diameter, ft. \text{34.4}

b. Active Height, ft. \text{23.6}

c. Active Core Volume, ft\textsuperscript{3} \text{20,400}

d. Total Uranium Loading, kg-U \text{222,000}

e. Average U-235 Content, \% by wt. \begin{align*}
&1.29 \text{ for } 2200 \text{ MWD/T} \\
&1.40 \text{ for } 5500 \text{ MWD/T} \\
&1.76 \text{ for } 11,000 \text{ MWD/T}
\end{align*}

f. Structural Material \text{Graphite}

g. Neutron Moderator \text{Graphite}

h. Moderator to Fuel Ratio

(1) For Magnesium Matrix \begin{align*}
&18.42 \text{ Volume C-H}_2\text{O} \\
&\frac{\text{Volume U-Mg}}{}
\end{align*}

(2) For 9\% Molybdenum-Uranium Alloy \begin{align*}
&38.50 \text{ Volume C-H}_2\text{O} \\
&\frac{\text{Volume Mo-U}}{}
\end{align*}

3. Reflector or Blanket

a. Material \text{Graphite}

b. Axial Thickness, ft. \text{Top 5.90, Bottom 4.59}

c. Radial Thickness, ft. \text{Sides 2.62}

4. Fuel Elements (for each type)

a. Fuel Material \begin{align*}
&\text{Metallic Uranium in Magnesium} \\
&\text{Matrix or 9\% Molybdenum-Uranium} \\
&\text{Alloy}
\end{align*}

b. Fuel Element Geometry \text{Tubular}

c. Clad Material \begin{align*}
&AISI 321 \text{ similar to Russian} \\
&\text{Alloy Type } 1X18H9T, 18\% \text{ Cr,} \\
&9\% \text{ Ni, and 1\% Ti.}
\end{align*}

d. Fuel "Meat" Thickness or Diameter, in.

(1) For Magnesium Matrix \text{0.946 OD - 0.236 Thick}
(2) For 9% Molybdenum-Uranium Alloy

<table>
<thead>
<tr>
<th>e. Clad Thickness, in.</th>
<th>0.765 OD, 0.126 Thick</th>
</tr>
</thead>
<tbody>
<tr>
<td>Russian Design Equivalent:</td>
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</tr>
<tr>
<td>In-flow central tube - .034&quot;</td>
<td></td>
</tr>
<tr>
<td>Fuel pressure tube - .020&quot;</td>
<td></td>
</tr>
<tr>
<td>Outer fuel cladding - .008&quot;</td>
<td></td>
</tr>
<tr>
<td>Code Alternative:</td>
<td></td>
</tr>
<tr>
<td>In-flow central tube - .050&quot;</td>
<td></td>
</tr>
<tr>
<td>Fuel pressure tube - .020&quot;</td>
<td></td>
</tr>
<tr>
<td>Outer fuel cladding - .008&quot;</td>
<td></td>
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</table>

| f. Fuel-Clad Gap (Cold), in. | 0 |

5. Fuel Assemblies (for each type)

<table>
<thead>
<tr>
<th>Steam Generating</th>
<th>Superheating</th>
</tr>
</thead>
<tbody>
<tr>
<td>a. Total Number</td>
<td>1,448</td>
</tr>
<tr>
<td>b. Number of Elements (Rods) per Assembly</td>
<td>6</td>
</tr>
<tr>
<td>c. Cross Sectional Dimensions, in.</td>
<td>Assembly OD-3.23&quot;</td>
</tr>
<tr>
<td></td>
<td>Cen. Tube ID-.825&quot;</td>
</tr>
<tr>
<td></td>
<td>Fuel Tubes ID-.434&quot;</td>
</tr>
<tr>
<td>d. Lattice Spacing, in.</td>
<td>7.876</td>
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<tr>
<td>e. End Fitting Materials</td>
<td>316</td>
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6. Reactor Control

<table>
<thead>
<tr>
<th>Control</th>
<th>Compensating</th>
<th>Shutdown</th>
</tr>
</thead>
<tbody>
<tr>
<td>a. Method of Control</td>
<td>Rods</td>
<td>Rods</td>
</tr>
<tr>
<td>b. Absorber Material</td>
<td>Boron-SS</td>
<td>Boron-SS</td>
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<tr>
<td>c. Number of Control Elements</td>
<td>24</td>
<td>162</td>
</tr>
<tr>
<td>d. Cross Sectional Dimensions in.</td>
<td>3-1/4</td>
<td>3-1/4</td>
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<tr>
<td>e. Effective Length, ft.</td>
<td>23.6</td>
<td>23.6</td>
</tr>
<tr>
<td>f. Type of Drive</td>
<td>Ball Screw</td>
<td>Ball Screw</td>
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### D. Performance Data

<table>
<thead>
<tr>
<th></th>
<th>Steam Generating</th>
<th>Superheating</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Reactor Coolant</td>
<td>Boiling Water</td>
<td>Steam</td>
</tr>
<tr>
<td>2. Reactor Coolant Outlet Temp., °F</td>
<td>644</td>
<td>950</td>
</tr>
<tr>
<td>3. Reactor Coolant Inlet Temp., °F</td>
<td>572</td>
<td>602</td>
</tr>
<tr>
<td>4. Primary System Operating Pressure, psig</td>
<td>2,200</td>
<td>1,570</td>
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<tr>
<td>5. Primary Coolant Flow, lb/hr</td>
<td>7.847x10⁶</td>
<td>2.80x10⁶</td>
</tr>
<tr>
<td>6. Avg. Core Coolant Velocity, ft/sec</td>
<td>7.8-25.3</td>
<td>75.5-154.1</td>
</tr>
<tr>
<td>7. Max. Fuel Center Temp., Mg Matrix, °F Mo Alloy, °F</td>
<td>745</td>
<td>1,080</td>
</tr>
<tr>
<td></td>
<td>725</td>
<td>1,050</td>
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<tr>
<td>8. Max. Cladding Temperature, Mg Matrix, °F Mo Alloy, °F</td>
<td>670</td>
<td>1,025</td>
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<tr>
<td></td>
<td>650</td>
<td>1,000</td>
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<tr>
<td>9. Burnout Heat Flux, Btu/hr-ft²-°F</td>
<td>373,000</td>
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<tr>
<td>10. Max. Core Heat Flux, Btu/hr-ft²-°F</td>
<td>194,000</td>
<td>177,000</td>
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<tr>
<td>11. Avg. Core Power Density, kwt/ft³</td>
<td>41.5</td>
<td></td>
</tr>
<tr>
<td>12. Peak to Average Power Ratio</td>
<td>1.1 to 1.4 radial</td>
<td></td>
</tr>
<tr>
<td></td>
<td>1.5 longitudinal</td>
<td></td>
</tr>
<tr>
<td>14. Fuel Management</td>
<td>(See following table)</td>
<td></td>
</tr>
<tr>
<td>15. Avg. Fuel Burnup, MWD/mt*</td>
<td>2,200, 5,500, or 11,000</td>
<td></td>
</tr>
<tr>
<td>16. Peak to Average Burnup Ratio</td>
<td>At 2200 MWD/T 1.9-2.04 (1st Load) At 5500 MWD/T 2.04(1st load)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>1.65 At 11,000 MWD/T 1.84(2nd Load)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>1.65</td>
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### E. Suggested Containment

<p>| | |</p>
<table>
<thead>
<tr>
<th></th>
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</tr>
</thead>
<tbody>
<tr>
<td>1. Design Criteria</td>
<td>1/2% per day leakage</td>
</tr>
<tr>
<td>2. Type</td>
<td>Vapor Suppression &amp; Containment</td>
</tr>
<tr>
<td>3. Primary Loop Coolant Inventory</td>
<td>185,200 lb. 126.4x10⁶ Btu</td>
</tr>
<tr>
<td>4. Secondary Loop Coolant Inventory in Sphere</td>
<td>72,300 lb. 67.1x10⁶ Btu</td>
</tr>
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</table>

*Metric tons are used throughout entire report except where noted.*
5. Geometry
   Sphere

6. Dimensions
   180' ID

7. Design Pressure
   17 psig

8. Material
   A-212 B
<table>
<thead>
<tr>
<th></th>
<th>Mg. Matrix - 0.020&quot;</th>
<th>Mg. Matrix - 0.040&quot;</th>
<th>Mo-Uranium Alloy - 0.020&quot;</th>
</tr>
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<tbody>
<tr>
<td><strong>Initial Enrichment (%235)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>1.28</td>
<td>1.41</td>
<td>1.41</td>
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<tr>
<td>Replacement</td>
<td>1.28</td>
<td>1.41</td>
<td>1.72</td>
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<tr>
<td><strong>Discharge Exposure (MWD/T1)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>2200</td>
<td>5500</td>
<td>2200</td>
</tr>
<tr>
<td>Replacement</td>
<td>2200</td>
<td>5500</td>
<td>11,000</td>
</tr>
<tr>
<td><strong>U-235 Burnout (g/T1)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>2440</td>
<td>5100</td>
<td>2440</td>
</tr>
<tr>
<td>Replacement</td>
<td>2440</td>
<td>5100</td>
<td>9270</td>
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<tr>
<td><strong>Total Pu Buildup (g/T1)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>1420</td>
<td>2740</td>
<td>1325</td>
</tr>
<tr>
<td>Replacement</td>
<td>1420</td>
<td>2740</td>
<td>4015</td>
</tr>
<tr>
<td><strong>Final U-235 Enrichment (gm 235 /initial g-U x 100)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>1.04</td>
<td>.89</td>
<td>1.32</td>
</tr>
<tr>
<td>Replacement</td>
<td>1.04</td>
<td>.89</td>
<td>.78</td>
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<tr>
<td><strong>Average Batch Fraction</strong></td>
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</tr>
<tr>
<td>First Load</td>
<td>1.00</td>
<td>.25</td>
<td>.25</td>
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<tr>
<td>Replacement</td>
<td>1.00</td>
<td>.25</td>
<td>.125</td>
</tr>
<tr>
<td><em><em>Time Between Batch Replacements</em>(yrs.)</em>*</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>1.98</td>
<td>1.24</td>
<td>1.24</td>
</tr>
<tr>
<td>Replacement</td>
<td>1.98</td>
<td>1.24</td>
<td>1.24</td>
</tr>
</tbody>
</table>

* At:
222,000 kg = 222 T = U inventory
848 x .8 x 365 = 2.476x10^5 MWD/Yr.
U-238 Burnout 1640 3440 3440
U-236 Buildup 370 815 815
U Loss Fraction .35% .77% .77%
SUMMARY OF BURNOUT-BUILDUP DATA

Mo-Uranium Alloy, Fuel Tube Wall .040"

<table>
<thead>
<tr>
<th></th>
<th>A&quot;</th>
<th>B&quot;</th>
<th>C&quot;</th>
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</thead>
<tbody>
<tr>
<td>Initial Enrichment</td>
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<td></td>
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<tr>
<td>% 235</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>1.71</td>
<td>1.86</td>
<td>1.86</td>
</tr>
<tr>
<td>Replacement</td>
<td>1.71</td>
<td>1.86</td>
<td>2.17</td>
</tr>
<tr>
<td>Discharge Exposure</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>MWD/T₁</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>2200</td>
<td>5500</td>
<td>5500</td>
</tr>
<tr>
<td>Replacement</td>
<td>2200</td>
<td>5500</td>
<td>11,000</td>
</tr>
<tr>
<td>U-235 Burnout</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>g/T₁</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>2490</td>
<td>5600</td>
<td>5600</td>
</tr>
<tr>
<td>Replacement</td>
<td>2390</td>
<td>5280</td>
<td>10,200</td>
</tr>
<tr>
<td>Total Pu Buildup</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>g/T₁</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>1290</td>
<td>2510</td>
<td>2510</td>
</tr>
<tr>
<td>Replacement</td>
<td>1290</td>
<td>2510</td>
<td>3850</td>
</tr>
<tr>
<td>Final U-235 Enrichment</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(gm 235 / initial g-U) x 100</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First Load</td>
<td>1.46</td>
<td>1.29</td>
<td>1.29</td>
</tr>
<tr>
<td>Replacement</td>
<td>1.46</td>
<td>1.29</td>
<td>1.13</td>
</tr>
</tbody>
</table>
IV. PLANT DESCRIPTION

The following descriptive material and drawings were prepared to fill out the design of the plant described without much detail in the Russian references, to permit better appraisal of the technology and to provide a basis for capital and operating cost estimates.

GENERAL BUILDING ARRANGEMENT

The general arrangement of the 300 MWe plant is shown in Figures 1 through 9. The 180 foot diameter sphere containing the reactor is flanked on one side by the fuel handling building and on the other side by the reactor service and turbine building. Other auxiliary structures such as the intake structure, storage and holdup tanks, unloading station, filter building, contaminated waste storage and drumming and administration building are located within the 550 x 670 foot reactor exclusion area. The switchgear area is located immediately adjacent to the exclusion area. The usual rail and road facilities are also shown on the plot plan. (The alternate plot plan shown in the Kaiser Study, KE 60-31 may be used.)

The reactor service building houses the turbine generators, the reactor control room, ventilation fans, water treatment, miscellaneous pumps, switchgear and other equipment required for operation. An instrument room is located directly below the reactor control room. This instrument room contains the terminations for all control signals, relays, wiring terminals and other control paraphernalia associated with the reactor and turbine. Such an arrangement simplifies the control room arrangement and centralizes all critical circuit components. The clothing change rooms and some offices are also located in the reactor service
building. One sphere access lock is located on the reactor service building side of the sphere and the other access lock is located on the fuel handling building side. The equipment access cover is also located on the fuel handling building side because of better access and installed crane facilities.

Inside the sphere the reactor is centrally located, with the top cover of the reactor at elevation + 84'-6" above the bottom of the sphere. On the right and left sides of the reactor are the equipment cells housing the headers, pumps, steam separators, evaporators, emergency condensers and preheaters. The equipment cells on each side of the reactor are connected by a pipe tunnel crossing at the front of the reactor. This tunnel is for service piping, interconnecting piping or cross ties and to provide a steam venting path to the left side of the reactor where the vapor suppression vents are located. The equipment cells will be designed to withstand internal pressures of 20 psig in order for the vapor suppression system to function properly and will be sealed to the extent that most of the steam that would be released in them would be vented through the two 6 foot vapor suppression ducts into the fuel discharge pool.

The thick cell walls serve two other purposes; shielding around the primary and secondary loops and blast protection for the sphere. The residual activity of the water will be low but the $N_{16}$ activity during operation will be high. The walls enclose all parts of the primary and secondary loop so they form an effective blast shield to protect the containment sphere from damage should a pipe or equipment shell fail. There are removable sections in the cell roof to facilitate the replacement of small equipment items such as pumps and valves.
The back of the sphere is filled with the fuel element storage area and discharge basin, the functions of which are described more fully in another section of this report. The front of the sphere has six general areas and functions. Two shielded rooms are provided; one for the recirculating rod coolant system equipment and the other for the reactor gas system equipment and primary loop cleanup bypass demineralizers. These were located on top in order to facilitate pump and equipment handling with the building crane.

Immediately below these equipment rooms are two instrument rooms for flow transducers and rupture detection equipment. The tunnel connecting the right and left side equipment cells and carrying the crosstie piping, service piping and drains is located at the front of the reactor. This tunnel has a branch that runs to the sphere for instrumentation, controls, steam and feedwater lines. This area also provides access and services to the control rod drive room located immediately below the reactor.

The bottom of the sphere is placed 50 feet below ground level to permit an optimum building layout. The irradiated fuel element storage basin must have a depth of 24 feet to provide shielding. Removal of the 46 foot long fuel element from the sphere and subsequent storage in the basin, requires a sphere base elevation of -50 feet. Placing the storage pit below ground level reduces shielding problems and facilitates fuel handling problems and cask car loading.

The reactor and sphere foundation will be designed for Zone I earthquake requirements. This will not add significantly to the cost of the plant.
CONTAINMENT

The Russian superheat reactor is located in the Ural mountains and no containment is provided other than that afforded by the shielding walls of the reactor hall. By contrast the study location is within 35 miles of a large western Massachusetts city, and the AEC ground rules require that some type of containment be included in the design. This is one of the major departures from the original Russian design.

The reactor container size is dependent on two factors: (1) The size of the reactor and equipment to be located inside the container, and (2) the space needed to charge and discharge reactor fuel elements. It was decided that the entire primary or boiling portion of the system (pumps, steam separators, evaporators and preheaters) would be located inside the container. Those portions of the secondary or superheating portion of the system inside the container would be the piping, emergency condenser and the secondary side of the preheaters and evaporators. This would permit emergency cooling of the reactor on a natural circulation basis by equipment located entirely within the container. The reactor fuel elements are 46 feet long over-all. It was assumed that the element would have to be assembled and disassembled outside the sphere. Thus, considerable clearance is required between the container and side and top of the reactor for fuel element handling.

It was found that a 180-foot diameter sphere would provide room for equipment installation and fuel element handling. The free volume in such a sphere, after the reactor vessel and equipment volumes are subtracted, is approximately 2.2 million cubic feet. This volume is more
than adequate to provide containment of the primary and secondary loop coolant volumes within the sphere if a 30 psig design pressure is used. However, considerable cost savings can be realized if a sphere of minimum wall thickness is used in conjunction with vapor suppression. Therefore, a 17 psig internal design pressure (-0.8 psig vacuum) has been assumed for the sphere. The total allowable leakage from the sphere would be less than 1/2 percent per day. The cells surrounding the major equipment and piping will be partially sealed and vented to the fuel discharge basin through two 6-foot diameter vents.

All vessels, equipment and piping will be separated from the sphere by thick reinforced shielding walls. This will provide some measure of internal blast protection for the sphere. In addition vacuum relief devices shall be provided to protect against pressures below 13.9 psia.

Any container must have penetrations to provide services while the reactor is operating. Some of these such as instrument lines, electrical controls, power cables and closed piping systems can be installed with permanent seals at the container barrier. Others, such as ventilation ducts, doors and open drains require the use of automatic closures. Isolation valves will be required in the superheated steam line to the turbine and in the feedwater line to the preheater. A cooling water supply and vent for the emergency dump condenser and other services will penetrate the container. The ventilation supply and exhaust ducts will be supplied with double closures to be actuated automatically on a building pressure increase. The open drain closures from the sphere shall also be provided with double automatic closures.
Equipment and personnel will enter the sphere by way of two air locks, each with a 4 x 8 foot opening. Large equipment will enter the sphere by way of a 10 x 10 foot opening in the sphere. This will be a bolted access plate with suitable seal. Two 3 foot diameter by 50 foot long locks are provided for moving fuel assemblies into and out of the sphere.

A maximum credible accident (an instantaneous circumferential rupture of the largest pipe in the cooling system) could result in the loss of all reactor coolant or disruption of natural circulation cooling. In this case, the decay heat of the reactor core, the reaction heat of any chemical combinations (i.e., magnesium and water) and all sensible heat inside the sphere must be dissipated by other means. Heat transfer through the sphere and its insulation is not sufficient to do this, so a fog spray system will be installed inside the sphere. A 1000 gpm system would be able to absorb up to 20 MW of heat. The pumps for the system would be electric driven from the backup diesel engine. The water supply would come from the 450,000 gallon fuel discharge pit inside the container.

**CONTROL AND SAFETY RODS**

A total of 241 reactor control rods, located as shown in Figure 9 are required. The total number of rods is large because of the large temperature coefficient of the reactor. Only 55 of the rods are required to shut down the reactor, but all are required to keep the reactor sub-critical during extended outages. The general rod design and mode of operation varies somewhat from that used in the original design. The main differences are the type of drives and the fact that all rods in the proposed design are driven from the bottom of the reactor. This
change was necessary because the lengthened core does not leave sufficient space above the core for scram rod withdrawal. Lengthening this space would require an equal increase in fuel element length with the resulting higher building costs and more difficult handling problems.

All rods are driven from the bottom of the reactor, although rod tips are replaced through the top of the reactor. A 3-1/2" OD x .065" wall thimble through the graphite moderator serves as a rod guide or operating channel and as a coolant channel. This thimble is of zirconium alloy and is sealed at both ends where it passes through the steel tank that contains the nitrogen atmosphere. The rod tips are of boron stainless steel.

A total of 24 control rods are required. These rods are driven by variable speed DC motors. Rod withdrawal speeds are limited to prevent fast reactor periods. The drive for the rod is a ball screw with suitable locks for holding the rod in position. The control rods are automatically inserted into the reactor on a scram signal. However, scram insertion speed is not critical and may be as long as one minute.

Fifty-five hydraulically driven scram rods are required for rapid shutdown of the reactor. A special latch holds the rods in the reactor; however, a slow acting ball screw follows the rod into the reactor to provide a positive lock to keep the rod in the reactor. These rods will shut down the reactor in less than two seconds and the ball screw will provide a positive lock in less than one minute. Accumulators (2 cu.ft. per rod) are required to insert these scram rods. The accumulators, one for each rod, are mounted on the walls of the rod drive room.
One-hundred sixty-two slow acting ball screw drives are inserted into the reactor on shutdown. These rods are required only for long shutdowns to compensate for the temperature effects and xenon decay. Therefore, their insertion time is not critical although they will be inserted automatically at every reactor shutdown.

All rods are water cooled with the supply at the bottom and a drain at the top. Actually the rod thimble forms a portion of the coolant channel as water will be introduced through the rod and be withdrawn from the top of the thimble. Because of coolant activation and waste disposal problems, a recirculating rod cooling system will be utilized. For normal operation approximately 2000 gpm of water will be circulated. The temperature rise for this water was assumed to be 70°F which results in a heat load of some 20.5 MW. The recirculating rod coolant is demineralized water, cooled in a heat exchanger (4500 sq. ft.) and circulated by two 2000 gpm, 100 psi head pumps. Raw water is used for cooling.

As a backup for the rods, a liquid poison system is installed. The poison material (sodium pentaborate) is stored in a 2000 gallon tank above the reactor. The poison system is manually actuated to introduce the material into the rod cooling system and fills the rod thimbles to effectively poison the reactor. The operating time for this system will be less than one minute.

REACTOR SHIELDS

The bottom shield of the reactor is a massive concrete slab which is pierced by the reactor rods. The rods and their cooling water provide adequate shielding for the 241 rod openings. This shielding must prevent activation of the rod drives during reactor operation and must
permit personnel access during shutdowns for maintenance purposes.
Some cooling in the form of embedded coils will be required in this bottom shield.

The side shields of the reactor consist of a water tank and concrete shielding walls. The water tank, approximately 3'-3" thick and 38'tall surrounds the reactor. This tank is filled with demineralized water kept clean by a small bleed-feed system. The tank is compartmented so a leak in any one section will not result in loss of the entire shield. This tank is cooled by coils that circulate raw water on a single-pass basis. The actuation of this coolant should be relatively low, however, if activation and disposal are a problem, a recirculating system could be used. Outside of this shield is a standard massive concrete shielding wall or cylinder. In locations such as the connector entry, where thickness is a problem, heavy aggregate concrete shielding is utilized.

The major shielding above the reactor consists of a 20-inch layer of cast iron that rests on the graphite. This shield is below the reactor atmosphere container and is pierced by all the fuel and rod channels. Special shielding plugs are installed in the rod channels at this elevation and each fuel assembly has a shielding section in this general area. The purpose of these top shields is to reduce neutron activation of the connectors and upper ends of the fuel assemblies. This portion of the top shield runs rather hot as its only cooling is by transfer to the fuel element coolant streams.

Immediately above the fuel assemblies is a reactor cover which actually forms the personnel shield above the reactor. This shield is made up in three layers. The lowest two layers are cast iron blocks that rest
on the fuel assembly heads. Above the cast iron is a steel cover plate. The reactor cover outside the fuel pattern rests on steel supports. Cooling is provided for the reactor cover.

**REACTOR VENTILATION**

The reactor sphere ventilation system consists of a supply unit, an exhaust unit and several local circulator-coolers. The supply unit will have a capacity of 20,000 cfm and provisions for filtering, preheating, washing and reheating before it is introduced into the sphere. Within the sphere local electric powered circulators with water cooling coils will be used for circulation and to remove heat. The ambient operating temperature in the sphere should be maintained at $120^\circ$F or less.

The exhaust from the sphere is by way of a fan, filter system and stack. During reactor operation and shutdowns, the pressure in the sphere will be slightly negative. All exhaust from the irradiated fuel storage, decontamination and disassembly areas (approximately 20,000 cfm) is by way of the filter system. Likewise the exhaust and purge from the reactor nitrogen system and the incinerator off-gas is by way of the filter system. The filters are designed to remove 99.9 percent of the particulates and 95 percent of the halogens in the exhaust stream. A double bank filter system is used. One bank handles the sphere exhaust stream and the other the exhaust stream from the fuel handling building.

The ventilation supply and exhaust ducts are equipped with automatic closures to effect containment sealing in case of a reactor incident. These closures will be automatically actuated on high activity levels in the exhaust air or a rise in pressure and humidity in the sphere.
REACTOR CORE

Figure 5 shows a cross section of the reactor core. Portions of the core and its appurtenances such as the foundation, shielding, rods, fuel elements and coolant connections are discussed elsewhere.

Immediately above the reactor foundation is the core support structure. The reactor atmosphere container sits on top of this support structure with the space between the foundation and the container filled with grout. Cooling tubes are attached to the support structure to aid in cooling. The graphite and fuel element support structure rests on the top surface of the reactor container bottom. Besides supporting and spacing the graphite, this structure supports and guides the bottom end of the fuel elements.

The graphite stack is a vertical cylinder using a square 7.876 inch lattice. There are a total of 1980 fuel channels and 241 rod channels plus miscellaneous instrumentation channels that run vertically through the graphite. The bottom reflector on the reactor is 4.6 feet thick, the active core is 23.6 feet high and the top reflector is 5.9 feet thick, giving an over-all height of the reactor stack of 34.1 feet. The active core is 34.12 feet in diameter (i.e., 52 lattice units) and is surrounded by a 2.62 foot thick reflector giving a total moderator diameter of 39.36 feet.

Reactor grade graphite with a density of 1.7 g/cc is used for the active core. AGOT grade graphite with a density of 1.7 g/cc can be used for the reflector. The total weight of unmachined graphite required for the moderator is 1850 tons. After machining, 1390 tons are left resulting in an effective density of 1.4 g/cc.
Purifying graphite bars 8 inches square will present no problems, and machining can be done according to standard practices. Tolerances for surfacing can be held within ± .002 inches, ± .006 inches for drilling and the cumulative tolerance across the reactor can be held within ± .045 inches by machining special blocks. The fuel element and control rod channel penetrations in the top and bottom of the reactor atmosphere container must line up with holes in the moderator. This is essential because of the limited clearance between the fuel element and the moderator. Spacing of the vertical graphite channels is maintained by (1) guide pins at the top and bottom of each channel, (2) by the control rod thimbles, (3) by the fuel elements themselves, and (4) by keys in the graphite stack. In addition, there are supports and guides between the side reflectors and the reactor atmosphere can to keep the core in place.

Two items that must be kept in mind during design of the graphite stack and its alignment devices are resistance to earthquakes and venting from ruptured fuel elements. The graphite stack and its foundations can be designed to resist without damage earthquakes with accelerations of 0.2 g. This is greater than the maximum earthquake intensity of 8 on the Rossi-Forel scale that has been recorded for the hypothetical reactor site.

Designing the moderator for venting a coolant tube rupture is a problem peculiar to tube type reactors and solid moderators. The rupture of one of the six coolant return channels in a fuel element may result in loss of the other five due to lack of adequate cooling.

There is no indication if any consideration was given this point in the original Russian superheating reactor design, but it was mentioned in the
recently received Vienna paper.\(^4\) It is felt that consideration of this point is necessary and that a venting system can be designed. Besides the graphite stack modifications, a duct would have to be run from the reactor gas container to the fuel discharge pool to act as a relief valve and an escape channel for the steam.

**REACTOR PIPING**

The reactor core has two coolant systems; a primary or steam generating system and a secondary or superheating system. These two coolant systems are shown schematically on Figure 2 with the proposed equipment and piping layout shown in Figures 7 and 8. There are a total of 1448 fuel element channels for steam generating and 532 fuel element channels utilized for superheating. The arrangement of these channels in the core is shown in Figure 9.

Four canned rotor pumps are used to recirculate the primary coolant. These pumps are manifolded at their suctions and discharge to permit reactor operation with one or more pumps out of service. Each pump has a stop check valve on its outlet and a shutoff valve on its inlet for maintenance purposes. Four 12-inch headers from the pumps feed twelve 6-inch distribution headers. Each of these headers has a flow control valve and feeds some 130 steam generating assemblies.

Individual connectors run from the supply headers to the steam generating fuel elements. Each of these connectors is a 1-1/4 inch OD x .125 inch wall 316 stainless steel tube and is provided with a shutoff valve and flow orifice at the takeoff from the header. The coupling between the connector and the fuel assembly has an automatic shutoff valve for fuel replacement. Similar connectors run from each steam generating element
The steam piping from the separator to the evaporators and the water piping from the evaporators and separators through the preheaters and to the pump suction is of stainless steel. The temperature range in the primary system is 572 F minimum and 644 F maximum. The maximum system pressure at the pump discharge is approximately 2200 psig. These operating conditions will require the use of schedule 160 pipe in sizes ranging from 6-inch to 14-inch. Stainless steel will be used to minimize corrosion and erosion problems encountered in handling the primary coolant. This entire primary system is located within the containment vessel.

The secondary or superheating coolant enters the sphere from the feedwater system. After passing through the four preheaters, it enters four double drum evaporators. Saturated steam from these evaporators is piped to four 10-inch headers. As in the primary system each header has a flow control valve and each connector has an isolation valve and flow orifice. The connector size and material are identical with those of the primary loop. From the fuel elements, the connectors lead to two 18-inch superheated steam collection headers. These headers join into one superheated steam line which leaves the containment sphere. Isolation valves are required at the sphere boundary on the superheated steam line and the feedwater supply line.

The secondary loop temperatures vary from 400 to 950 F with pressure ranges from 1600 to 1300 psia. The piping material is 1-1/4"chromium steel; similar to standard power plant practice for this pressure and temperature range. The pipe diameters as shown on Figure 2 vary from 8 to 24 inches, with wall thickness in the schedule 120 to 160 range.
The connectors to and from the fuel in both the primary and secondary system are a major cost item. These 1-inch ID x 1-1/4-inch OD connectors of 316 stainless steel average 95 feet in length for a total of approximately 72 miles of tubing. These connectors have to be continuous over their length from the flow orifice to the fuel element.

Routing of the tubing above the reactor presents problems in space utilization, thermal expansion, stresses, and connections. It is proposed that the connectors be routed two wide and 26 layers deep between adjacent rows of standpipes. Rod coolant return tubes would be routed through the same space. With this routing scheme, it is important that all control rods or other core penetrations be laid out so a clear path exists from side to side on the reactor between the standpipes. To the problems in routing as described above must be added the necessity of insulating between adjacent connectors. In the primary system, the temperature differences are minor but this is not true of the secondary system. The superheated steam return must be insulated to prevent heat loss to adjacent connectors and rod coolant return lines.

The primary and secondary coolant supply manifolds are approximately 70 feet long. Since there are a multitude of individual lines, insulation of these lines is impractical. Therefore, the entire header-connector area is insulated. As can be seen in Figure 7, the headers are spaced to provide an access way for maintenance of the valves, fittings and instrumentation.

Both the primary and secondary loops will require special construction procedures emphasizing quality of materials, workmanship and inspection and extreme cleanliness. Special cleaning procedures will have to be followed during construction.
The volume of the primary loop is 7200 cubic feet, giving a recirculation time of approximately 85 seconds. The pressure drop in this loop is 140 psi. The volume of the secondary loop inside the reactor sphere is 7000 cubic feet. The values of pressure and temperature shown in Figure 3 are at the outlet of the active section of the superheating fuel elements. Most of the pressure drop to the turbine throttle valve is accounted for in the fuel element and connectors before the steam enters the main header.

Wherever possible, pressurized drains will be used from the sphere. Normally clean drain will be routed to the effluent canal. Normally contaminated drains will be routed to the waste holdup tanks. Open drains that are below elevation +40 feet will run to a sump where they will be pumped to the waste holdup tanks.

**REACTOR COOLANT SYSTEM EQUIPMENT**

**Primary Circulation Pumps**

Four pumps are provided to circulate the primary coolant through the reactor. The pumps are of the canned rotor type requiring no shaft seals.

**Specifications**

<table>
<thead>
<tr>
<th>Specification</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Material</td>
<td>All in contact with pumped fluid, 304 stainless steel</td>
</tr>
<tr>
<td>Flow per Pump</td>
<td>5,900 gpm</td>
</tr>
<tr>
<td>TDH psi</td>
<td>140</td>
</tr>
<tr>
<td>Suction Temperature, °F</td>
<td>572</td>
</tr>
<tr>
<td>Suction Pressure, psia</td>
<td>2,100</td>
</tr>
<tr>
<td>Minimum NPSH</td>
<td>40 ft.</td>
</tr>
</tbody>
</table>

**Emergency Condenser**

Two emergency condensers are provided to remove reactor decay heat after a scram in the event the superheated steam line isolation valve is closed. Each condenser consists of a U-tube bundle located near the bottom of a horizontal cylindrical shell. The top of the bundle is
connected to the superheat steam line and the bottom drained to the boiling side of the evaporators. The shell is normally about 3/4 filled with demineralized water at ambient temperature and the tube side drain line valve closed. The tube side will fill with condensed steam. To put the condenser in operation, the drain valve will be opened, draining the tube bundle, admitting steam which will condense and drain to the evaporator.

The steam generated on the shell side at atmospheric pressure will be vented outside the containment sphere.

**Specifications**

**Tube Side**

| Inlet Flow, lb/hr. | 100,000 |
| Temperature, F | 950 |
| Pressure, psia | 1,565 |

| Drain Flow - water at saturated temperature - lb/hr. | 100,000 |

**Shell Side**

| Evaporation Rate, lb/hr. | 57,000 |
| Temperature, F | 228 |
| Pressure, psia | 20 |

**Separator**

The steam water mixture from the reactor flows to the two separators. The separated steam is discharged to the evaporators and the water drains to the preheaters.
Specifications

Material - all in contact with the steam water mixture or steam 304 s/s

Inlet Fluid - steam water mixture

Flow, lb/hr. 3,940,000
Quality, % 31.2
Pressure, psia 2130
Temperature, °F 644

Steam Discharge

Flow, lb/hr. 1,235,000
Quality, % 99.5

Drain - Solid Water

Flow, lb/hr. 2,705,000
Enthalpy, Btu/lb. 683.6
Maximum Pressure Drop, psi 10

EVAPORATOR

The four evaporators condense the steam discharged from the separators transferring heat to the steam water mixture from the preheater to supply 99.9 percent quality steam to the superheater. The flow diagrams in the Russian documents show the steam from the separators condensing inside the tubes with the boiling water on the shell side. A heat exchanger condensing steam outside the tube and boiling inside may be a better design.

Specifications

Condensing Steam Side

Fluid - condensing steam

Material - all in contact with fluid 304 s/s
Flow, lb/hr. 620,000
Pressure, operating psia 2,120
Inlet Quality, % 99.5
Drain - Enthalpy, Btu/lb. 683.6
Maximum Pressure Drop, psi 10

Boiling Side
Material - carbon steel
Flow in - steam water mixture, lb/hr. 702,000
- quality, % 27.4
Pressure - operating psia 1,570
- design psia 1,800
Flow Out - steam, lb/hr. 700,000
- quality, % 99.9
Blowdown flow, lb/hr. 3,500
Maximum Pressure Drop, psi 10

Preheater
The drains from the steam separator and evaporator flow through the tube side of the four preheaters. Heat is transferred from the tube side to heat and partially boil the feedwater.

Specifications

Tube Side
Fluid - demineralized water
Material - all in contact with fluid, 304 s/s
Flow, lb/hr. 1,970,000
Temperature, in, F 644
out, F 572
Pressure - operating psia 2,100
- design psia 2,400
Maximum Pressure Drop, psi 10

Shell Side
Fluid - feedwater and boiling water
Material - carbon steel
Flow, lb/hr. 702,000
Temperature, in - F 480
Discharge Flow Quality, % 27.4
Pressure - operating psia 1,565
- design psia 1,800

COOLANT SUPPLY AND TREATMENT

Main Cooling Water (No Contamination)
Raw river water will be used without normal treatment and returned directly to the river with no treatment.

Service and Sanitary Water
Well water will be provided without treatment for service and without treatment except for chlorination for sanitary usage.

The required flow rate of the well has been estimated as

Service Water 500 gpm
Sanitary Water 150 gpm
Storage Basins 1000 gpm
Demineralizer Influent 50 gpm 1700 gpm
A storage tank of 100,000 gallon capacity should be provided for service water. A 5000 gallon tank should be sufficient for sanitary water storage.

**Demineralizer Plant**

A three-bed demineralizer plant with vacuum degasifier is provided with a normal capacity of 50 gpm. This is broken down as:

- **Primary System Makeup** 10 gpm
- **Secondary System Makeup** 25 gpm
- **Rod Cooling System Makeup** 1 gpm
- **Shield Water Makeup** 5 gpm
- **Demineralizer Plant Usage** 9 gpm
  
  50 gpm

Piece No. 1 is a pair of rubber-lined cation demineralizers four feet in diameter by eight feet high. Each contains 50 cubic feet of hydrogen form cation resin.

Piece No. 2 is a 100 gpm vacuum degasifier capable of removing oxygen down to 0.005 cc/liter. It must be rubber lined or stainless steel to handle low pH water.

Piece No. 3 is a pair of 50 gpm centrifugal pumps. These must be stainless steel, rubber lined, or acid resisting bronze.

Piece No. 4 is a pair of anion exchangers very similar to Piece No. 1. Each contains 50 cubic feet of hydroxyl form anion resin.

Piece No. 5 is a pair of mixed bed ion exchangers, each approximately three
feet in diameter by seven feet high. Each contains approximately 25 cubic feet of mixed bed resin consisting of equal mole percents of hydrogen form and hydroxyl form resins.

Auxiliary equipment not shown would include the regeneration equipment, the controls, and the interconnecting piping. Also included would be a sulfuric acid storage tank and a 50 percent caustic storage tank of approximately 5000 gallons each with associated pumps and piping.

This plant should discharge into a 10,000 gallon stainless steel tank. The tank must have a floating roof or membrane to prevent pickup of gas from the air in the tank.

**Primary Reactor Cooling System (Boiling System)**

The water in the primary system will be maintained as high purity water with no additives. Purity will be maintained by a 100 gpm capacity bypass demineralizer. Only one unit will be provided with provisions to discharge spent resin and recharge fresh during reactor operation. The equipment would consist of:

1. The primary cooling system pumps. These provide the head necessary to drive the 100 gpm through this system.

2. A 200 square foot stainless steel regenerative heat exchanger.
   
   Design pressure is 2400 psia and design temperature is 650 F for both shell side and tube side.

3. A 300 square foot heat exchanger with stainless steel tubes at the above design conditions. The shell side carries river water for cooling.
4. A two-foot diameter by six-foot high stainless steel demineralizer tank. (Design conditions as above.) It would contain 15 cubic feet of ion exchange resins.

The only additional primary water treatment is to provide a vent from the high point of the steam system to provide a degas function. The flow should amount to less than 0.1 gpm.

**Secondary System**

The only water quality treatment for the steam cycle would be a 25 gpm blowdown from the boiler. The degas function would be accomplished in the main condenser and the degasifier. No additives would be used in the system.

**Rod Cooling System**

The control rods are cooled by a recirculating high purity system. Water purity is maintained by a 5 gpm bypass cleanup system. This is similar to the primary cleanup system in nature, but considerably simpler since the pressure and temperature would be low. The only requirements are a one cubic foot ion exchanger made from a 6-foot length of 6-inch pipe. The unit is connected across the main system recirculating pump, and normally contains one cubic foot of ammonium form ion exchange resin. This reduces corrosion in the system and permits the use of carbon steel piping. Occasional venting from the system high point provides sufficient system degas.

**Shield Cooling System**

The shield cooling system consists of coils located in the shield tank. These coils are cooled with untreated river water. Approximately 5000 square feet of coil surface is required.
Shielding Water

In order to reduce activity problems, the water in the shield tank will be kept at fairly high purity. This is accomplished by bleeding 5 gpm of the fluid to radioactive waste while an equal amount of high purity water is injected. It would be possible to add corrosion inhibitors to this system since the radiation levels are comparatively low.

DECONTAMINATION

Small Parts Decontamination

A small parts decontamination facility is provided for cleaning of pump parts, valve parts, etc., to ease maintenance problems. The facility consists of three heated and agitated stainless steel tanks and one carbon steel cold water rinse tank. Each tank is of about 400 gallon capacity.

Reactor Decontamination

At the present time, there is no satisfactory process for decontaminating a boiling water reactor. The corrosion film laid down by the oxygen-bearing water is not removed by the decontaminants being developed for pressurized water reactors. A research and development program would be required before a system decontamination facility could be designed.

The only features that could be designed into the reactor system to aid future decontamination would be:

1. Design for as near complete drainage and venting of the system as possible to facilitate rapid emptying and filling.

2. Allow adequate space as near the reactor as possible for future construction of a decontamination chemical handling and mixing facility.
REACTOR GAS SYSTEM

The reactor atmosphere can have an inside diameter of 39.4 feet and is 1 inch thick. A low alloy steel plate (SA-301, Grade B-1Cr-1/2 Mo) is required because the vessel temperature may be as high as 1100 F. In order to withstand the 5 psig internal pressure that might result from a fuel element failure, a 1-inch wall thickness is required. The vessel walls are 36.7 feet high. Some problems may be encountered in designing the flexible top connection for 5 psig internal pressure.

Spacing of the standpipes in the top cover is critical because these holes must line up with the holes in the graphite. An over-all tolerance in the range of ± 0.015 inches will probably be required.

Reactor Gas

Nitrogen is circulated through the reactor moderator at a normal rate of 400 cfm to provide:

1. Drying of the moderator following a fuel element rupture.

2. Fuel element rupture detection.

3. Preventing in-leakage of air and contaminants into the reactor by maintaining a slight positive pressure.

4. Controlling moderator temperature by controlling the thermal conductivity of gaps between moderator blocks.

If the moderator gets wet, the gas circulation rate will be increased to 1000 cfm to increase the drying rate.

The recirculating gas system includes a condenser, blower, precoolers, drying towers, regenerative condensers and filters. This equipment is located in a cell adjacent to the reactor. Tanks for storage, transfer
pumps, pressure control valves and an unloading station are provided outside the sphere. These components are part of the reactor gas storage and makeup system.

The nitrogen atmosphere around the moderator is contained by a steel tank enclosing the sides and bottom of the graphite stack. On top of the graphite stack is a cover, connected to the sides of the tank by a flexible diaphragm. Each fuel channel and control rod has a standpipe approximately 10 feet long welded to the top cover. The seal between the fuel element and the standpipe is of the labyrinth type and is formed by rings on the fuel element.

The reactor gas inlet is at the bottom of the reactor and the outlet at the top. Provisions are made for purging the recirculating gas system. All gas to be discharged is released into the ventilation exhaust system upstream of the filter system.

Nitrogen is suitable for use as a reactor gas and its compatibility with graphite has been demonstrated. However, there is the possibility that disassociated nitrogen, produced in the reactor, would tend to nitride the metallic components in the core. Such items as coolant tubes, rod thimbles and the tank would be subject to nitriding and its resulting embrittlement. Since some of these items remain in the reactor for extended periods, even very low rates of nitriding may not be acceptable.

The use of alternate reactor atmospheres should be studied. Helium in particular, with its higher thermal conductivity, should be looked at as one possible way of increasing the fuel element-graphite moderator annular clearance. The primary deterrent to the use of helium in the USA would be its higher cost.
FUEL HANDLING

Unirradiated fuel assemblies (see Figure 4) will be unloaded and inspected in the fuel handling building. The inspection will include examination for damage incurred during shipment and unloading, leak testing, orifice installation, marking and testing of any special fuel assemblies or instrumentation. After inspection the fuel assemblies will be stacked in a special container on the unirradiated fuel transfer cart. When the cart is loaded, it will be inserted into the unirradiated fuel transfer lock and thence into the reactor sphere. Inside the sphere, the special container will be raised to a vertical position by the building crane and the fuel will be stored in the space provided. Unirradiated fuel inspection and storage will take place during reactor operation.

The overhead crane will be used to remove the steel cover and cast iron covers above any fuel assembly that is to be replaced. Unirradiated fuel elements will be picked up from the fuel storage area by the overhead crane and transported to the charge-discharge machine. The fuel transfer machine will be positioned over the fuel channel to be recharged, will remove the last cast iron cover plate, will disconnect the coolant supply and return connectors on an irradiated element and then raise the element into a shielding cask. The new element will be lowered into place and the coolant connections made. The charge-discharge machine will then transport the irradiated fuel to the water pit. In this pit, a special tilting device will transfer the fuel element to a horizontal position on a transfer cart, which will move the irradiated fuel from the sphere to the storage basin by way of the underwater irradiated fuel lock.
All charging and discharging will be done when the reactor is shut down. The charge-discharge machine will be used to transfer irradiated fuel elements between reactor locations. The charge-discharge machine may also be required to ream or enlarge graphite channels that have contracted or been damaged. Special ventilation connections will be required for handling ruptured fuel elements. Provisions will be made for water cooling the irradiated fuel elements.

The design of the charge-discharge machine varies from the Russian design because a cask is required. The building layout inside the containment sphere does not provide any shielding above the reactor thereby requiring that the fuel be drawn up into a cask. By contrast the Russian design provides for shielding in the walls and roof of the reactor hall, but their design neglects containment.

Handling, charging and discharging of a fuel assembly 3.23 inch OD and 46 feet long will be difficult. One point of major concern is discharge of irradiated fuel with only a nominal .040 inches clearance on the diameter. Problems of graphite stack alignment, fuel element alignment and graphite contraction over extended exposure periods will undoubtedly complicate fuel replacement. Every effort should be made, consistent with heat transfer requirements for the moderator, to enlarge this clearance.

The fuel element cross section indicates some thin graphite sections. It is possible that some of these thin sections might fail and permit a piece of graphite to become loose. This piece could then jam and prevent removal of the fuel element. Since the only structural tensile members in the fuel element are the seven stainless steel tubes, pulling forces for fuel element removal will be limited to a few thousand pounds.
An area is provided in the fuel handling building for storage of irradiated fuel elements and their subsequent disassembly. Loading of the fuel into casks and loading of the casks into railroad cars for transportation to the separations facility will be done after a sufficient period of decay. As discussed elsewhere the fuel will be separated from the graphite and only the coolant return tube (fuel and cladding - total length about 24 feet by 1 inch OD) will be sent to the fuel processing plant.

Space has been provided in the fuel handling facility for decontaminating and reclaiming the end pieces and graphite. It was assumed that 75 percent of the top end pieces would be reclaimed and shipped to the fuel assembly plant. This mode of operation was assumed to minimize fuel costs. However, a more complete fuel analysis based on (1) end piece cost, (2) reclamation costs, (3) manpower requirements, (4) reassembly costs, (5) fuel life, and (6) waste disposal costs and other factors would have to be made before the final fuel cycle decisions were made.

**LIQUID WASTE DISPOSAL**

All liquid wastes which are suspected of radioactive contamination will be classified into high level and low level. The exact break point would be somewhat arbitrary.

**Low Level Waste Tanks**

The low level wastes are sent to a set of large holdup tanks. In most cases these tanks could be emptied into the condenser cooling water effluent for dilution and dispersion into the river after checking the contents for contamination level.

These low level tanks have a secondary function which would be to accept the coolant from both of the reactor cooling systems should they ever need
draining. Although these could not be called low level wastes, the low level waste tanks would provide a place to store the coolants until means of disposal could be provided.

The low level tanks would be four 25,000 gallon tanks, buried just below grade for shielding purposes. They should have light-weight roofs.

Pumping facilities will be provided to send the contents from any of these tanks to the high level waste tanks if their activity exceeds releasable limits.

**High Level Waste Tanks**

High level wastes are sent to another set of smaller tanks. These will be four 5000 gallon underground tanks. These wastes would require more extensive treatment than dilution before release to the river.

Three mixed bed demineralizers are provided in series to treat these wastes. The first bed in the series will remove over 99 percent of the contamination from a waste stream. The second bed is a safety in event of unexpected breakthrough. The third bed is available for use as a safety while the first bed is being recharged with fresh resin. The three beds would be rotated, the most recently recharged unit would be the last in line.

These beds have a nominal 50 gpm capacity, although they could be loaded up to 150 gpm capacity when needed.

Because of their non-regenerating mixed bed design, they are four foot diameter by four foot straight height carbon steel vessels with rather simple internals.
Each unit contains about 25 cubic feet of hydrogen form cation resin and hydroxyl form anion resin.

These units will become quite radioactive during use and therefore require shielding and remote operation.

**Storage Basin Overflows**

One of the largest radioactive liquid waste streams to be handled will be the overflow from the fuel element storage basins. This 1000 gpm flow will be required only when a large number of recently discharged fuel elements are being stored; normally the flow will be considerably less than this.

This stream will be sent to one of the low level liquid waste tanks for continuous monitoring. It is expected that the stream will nearly always be clean enough to send directly to river.

**Gaseous Waste Disposal**

This system is a relatively simple one. All points which could contain radioactive gases are piped to a central holding tank of perhaps 500 gallon size. Here any entrained moisture would separate out.

A positive displacement compressor draws gas from this tank and sends it to one of several (perhaps four) holding tanks. These hold the gas at about 100 psig and might be some 3000 gallons each.

The gas is stored in these tanks and monitored until the activity has decayed enough to allow dilution and dispersion. The gas could be discharged into the main exhaust tank.

It should be noted that the radioactive gas will usually be isotopes of hydrogen, oxygen, and nitrogen with short half lives. Only in the event of release of fission product gases would long term storage be required.
Combustible solid wastes such as contaminated paper and the used ion exchange resin can be burned in a special incinerator to decrease their volume and weight before drumming. The incinerator must have a carefully designed smoke stack to prevent radioactive ashes from being ejected; possibly this can be connected into the area filter and exhaust system. Sufficient pre-burning controls must be established to assure that there are no volatile radioactive materials in waste to be burned.

**REACTOR INSTRUMENTATION**

**Flow Instrumentation**

1. **Bulk Flow Instrumentation**

   Water and steam flow is measured by determining differential pressure across orifices or flow nozzles. Recorders are provided in the control room with adjustable trip points to cause low flow annunciation or if desirable, a reactor scram. Specific measurements include:

   a. Inlet bulk water flow rate to the reactor downstream from primary coolant pumps - 4 points.

   b. Inlet feedwater flow into each preheater - 4 points.

   c. Superheat steam flow out of steam collectors - 2 points.

2. **Individual Tube Flow Instrumentation**

   Individual tube flow monitors are not indicated in the Russian design, and not included in the cost estimates of this report. Although they would be a desirable operating adjunct they may not be needed if the other instrumentation systems operate successfully.
Temperature Instrumentation

Temperatures will be monitored at various places within the reactor coolant system as necessary to provide information to the reactor operators and to insure safe operation of the system.

Temperatures of the graphite moderator, the water shielding, and of the concrete shielding and reactor structural walls will be monitored as necessary. Process coolant temperatures will generally be measured with immersion type resistance temperature detectors or possibly in the superheated steam system by iron-constantan thermocouples inside stainless steel wells.

Graphite temperatures and concrete shield temperatures will be measured with chromel-alumel thermocouples enclosed in magnesium oxide and a stainless steel sheath.

Specific temperature measurements anticipated include:

1. Process Coolant Temperatures
   a. Inlet bulk water temperature - 4 points.
   b. All coolant temperatures entering and leaving the preheater - 4 points.
   c. Superheated steam temperature in each steam collector - 2 points.

   Two recorders are provided in the reactor control room to record all of the above temperatures. In addition alarm contacts are provided to annunciate out-of-limit conditions. Excessive superheat steam temperatures are used to initiate a reactor scram.

2. Moderator and Shield Temperature Monitoring
   a. Graphite temperatures are monitored at 25 locations within the reactor. Five vertical removable stringers are used, each containing 5 thermocouples spaced throughout the moderator.
b. Water shield temperatures are monitored at 12 locations at various points in the tank.

c. Exterior concrete shielding is monitored at 12 selected locations in the periphery. Six additional thermocouples are located in the concrete membrane supporting the core.

**Pressure Instrumentation**

Process cooling system pressures are measured with pressure measuring instrumentation at the following locations:

a. Inlet header, primary coolant system - 1 point.

b. Water downstream from primary coolant pumps - 4 points.

c. Steam separator pressure - 2 points.

d. Evaporator pressure, secondary side - 2 points.

e. Superheater steam collector pressure - 2 points.

Out-of-limit pressure trips are provided to cause annunciation and/or reactor scram.

**Nuclear Instrumentation**

Nuclear instrumentation is provided to monitor the complete flux range from that existing in the shutdown reactor to that equivalent to 150 percent of design power level. Complete coverage of this range requires three specific groups of instruments covering the subcritical, the intermediate and the operating flux range. Both flux level and period or rate of rise information are provided to the reactor operator.

Trip points are provided to cause annunciation and/or reactor scram whenever specified out-of-limit conditions exist. Specific nuclear instrumentation includes:
1. **Subcritical Flux Monitors**

Two channels of subcritical nuclear instrumentation are provided. Each channel will include a fission chamber located in a water-cooled thimble extending into the active zone from the (bottom) of the reactor. Amplifiers and log count rate meters are used to provide both period and flux level information throughout the subcritical region.

2. **Intermediate Level Flux Monitors**

Three channels of intermediate level instrumentation consisting of thermal neutron sensitive ion chambers are provided in penetrations from the bottom of the reactor. The ion chambers are connected to both log type current amplifiers and to period amplifiers to provide both decade flux level and period logic to the reactor operators. Trip points are provided to cause a scram on fast period indication. A trip will be caused by periods faster than 20 seconds.

3. **High Level Nuclear Instrumentation**

Flux levels in the power range (1% to 150% of design power) are monitored by two groups of four thermal neutron sensitive, uncompensated ion chambers. One group of four chambers is located in horizontal thimbles which penetrate the shielding to the edge of the active zone about 4 feet from the top of the active zone. The other group of four are in a similar ring about four feet from the bottom. Each channel is connected to current amplifiers with high level trips. The two groups are connected so that a reactor scram will be caused by a high flux trip in any two out of four combinations. Removal of one amplifier in either group will still leave three trip channels, any two of which will shut down the reactor.
In addition, three summing transducers are provided, each receiving signals from two of the high level chambers. The three signals from the summing transducers are used as input signals in three channels of power rate of rise instrumentation. Reactor scram signals will be caused by excessive rate of rise indication from any two of these instruments.

Fuel Element Failure Detection Instrumentation

The mechanism of rupture detection employed by the Russians is vaguely described as observing a pressure buildup in the gas space around the fuel elements. Further verification is obtained by monitoring the gas from selected areas for radioactivity. This scheme seems to be based on the assumption that any penetration of the inner cladding by the coolant stream will cause a reaction with the U-Mg fuel of such a violent nature that the outer cladding will also be quickly penetrated. This scheme will be maintained in the proposed adaptation. However, it is also considered desirable to provide indication within the coolant systems of any gross amount of contamination. One method of doing this would be to take samples of the steam phase from both the separators and the superheat collectors, cool, condense, and then pull any non-condensables through scintillation counters and monitor for radioactive noble gas fission products, i.e., xenon krypton. Samples of gas from the air ejector exhaust could also be monitored for similar activity. High activity signals or other evidence of a fuel element failure would cause an alarm; and the operators would manually scram or shut down the reactor.

Reactor Plant Control

An analysis of the dynamic control characteristics of the Russian design has not been attempted because of time limitations. The Russians imply that an
automatic rod control system is provided. However, because of a lack of analysis and a lack of operating experience with a similar concept, it is assumed the reactor would be started up under manual control as a base load station. The control scheme for the operating plant is shown on Figure A and discussed below.

1. Superheat Loop

The superheat loop will be operated nominally as a constant pressure, constant temperature system. The turbine inlet valves will be controlled to maintain constant upstream pressure. Turbine speed will essentially be controlled by line frequency, thus the turbine generator will not be a demand unit but will deliver as much power as possible up to a set maximum.

A turbine speed controller will, however, be provided to override the pressure control in the event of excessive electrical system fluctuations. If this causes the pressure in the superheat loop to increase past a certain point, the bypass valve will open permitting steam to flow directly through a desuperheater to the condenser.

A reactor scram will also be initiated by excessive pressure in the superheat system. However, the trip point to cause a reactor scram will be set sufficiently high to prevent minor line disturbances from causing a scram.

The superheating regulator described in the Russian design is retained as an ingenious way of maintaining constant outlet superheat temperature. A high superheat temperature causes the valve on the steam line to the superheating regulator to open. This causes an increase in the secondary feedwater temperature and therefore causes more pounds of saturated
steam to be generated in the evaporators. The net result is more steam flow through the superheat elements and therefore a lower outlet temperature.

Because of the long time delays involved the response rate of the controller must be fairly slow to prevent hunting tendencies. Therefore, the valve control system cannot be designed for very fast response. However, with proper design the superheating regulator provides a method of automatically compensating for the inevitable variations in relative power level between the superheat and boiling zones.

2. **Primary Loop**

The primary loop would be operated at a constant flow rate and with a controlled liquid level in the separator. Liquid level in the separator would be adjusted by control of the makeup rate.

3. **Automatic Operation of the Plant**

Analog studies of the dynamic operation of the plant and probably some initial operating experience would be necessary to define an optimum method of automatic control. However, there is no reason to believe that a control system cannot be designed to make the plant capable of automatically following electrical system demand.

The design of the lattice so the system is not strongly susceptible to changes in water density, and the existence of a fairly large negative metal temperature coefficient indicates an inherent ability of the reactor to respond to turbine demand. For example, if the system were designed for demand operation, an increase in turbine-generator load would cause the turbine inlet valve to open dropping pressure in the
superheat system. This would immediately be reflected in the evaporator dropping both the boiling pressure on the superheat side and the condensing pressure on the primary side. Subsequently the separator pressure would drop. This would be reflected by increased voids in the reactor but also by a lower boiling temperature. Since the lower temperature is the predominate effect the reactor power level would increase to match the higher load demand.

If the inherent increase in power level did not match the load demand, automatic adjustment of some control rods could be used to further increase the power level until the pressure in the superheat region had been restored to its control point.

One of the main areas of uncertainty is the relative inherent response between the superheat and the boiling regions. However, the superheating regulator would automatically adjust superheat flow rate for some difference in response rate.

TURBINE-GENERATOR, CONDENSER AND FEEDWATER SYSTEMS

Steam from the superheating section of the reactor will flow directly to the turbine-generator described in KE-60-31.*

A steam bypass is provided from ahead of the turbine stop valves directly to the condenser for use when the steam flow from the reactor is greater than required by the turbine-generator load demand. It would be used only on rapid reduction in generator load.

The steam flow to the turbine and bypass system is controlled by a combined throttle pressure and turbine load control governor. Normally the flow to

* op.cit. p. 3.
the turbine is controlled by the turbine load governor. The pressure regulator will override the load control if the steam pressure drops more than a nominal amount and will open the bypass to the condenser if the pressure increases.

The reactor feedwater is heated in six closed feedwater heaters by steam extracted from the turbine. The three high pressure heater drains are pumped to the feedwater pump discharge. The three low pressure heaters drain to the condenser hot well. The condensed turbine exhaust flow, heater drains, and system makeup are deaerated in the condenser.

The feedwater from the top extraction feedwater heater flows to the superheat regulator. This is an additional feedwater heater using throttle steam and is drained to the high pressure extraction feedwater heater. The steam flow is controlled by the superheater outlet temperature.
V. COMMENTS ON TECHNOLOGY AND ENGINEERING

A. PHYSICS CHARACTERISTICS

General

Physics characteristics of the Russian Beloyarsk reactor, as modified for 300 MWe operation, are summarized in the accompanying tables.

Two noteworthy characteristics of this reactor are:

(1) The reactivity is insensitive to void fraction or coolant density, and

(2) The neutron economy is exceptionally good for a stainless steel pressure-tube reactor.

These characteristics are discussed in subsequent sections of this review; two points brought out in the discussion are emphasized here because they are characteristic of graphite moderated pressure-tube reactors in general.

a. Because of the low neutron absorption cross section of graphite, the sacrifice in neutron economy required to make the lattice insensitive to water volume is small relative to that of a water-moderated reactor.

b. The use of pressure tubes requires substantial quantities of structural material in the lattice. If this material must be a strong neutron absorber, such as steel, good neutron economy can be maintained only at a sacrifice in terms of heat transfer area or fuel complexity or both. On the other hand, development of low absorbing materials adequate to the service conditions would permit the use of relatively simple fuel geometries with good heat transfer characteristics without greatly reducing neutron economy.
Neutron Economy

The pressure-tube and fuel cladding material for the reactor is stainless steel which has a relatively high neutron absorption cross section. In order to keep parasitic absorption in the steel to a minimum, the Russians have placed their pressure tubes in low thermal flux locations shadowed by the fuel and have used a very thin steel cladding on the outer fuel surface. In the modified 300 MWe design, the steel volume is only 12 percent of the fuel volume and neutron absorption in the steel is only about 8.5 percent of the neutron production. Since the water volume is also fairly small (45 percent of fuel volume), the over-all neutron economy of the lattice is high. This factor permits operation of the reactor with an enrichment of only 1.3 percent for a fuel discharge exposure of 2200 MWD/T. It should be noted, however, that the high neutron economy has been achieved only by accepting a low heat transfer surface area, which limits the specific power, and by designing a complex fuel assembly with internal fuel cooling. An optimum design would require a compromise between neutron economy and fuel cooling area or fuel complexity. It is not known how the present design would compare with such an optimum.

The use of nitrogen as the stack atmosphere contributes slightly to parasitic neutron absorption in the lattice. A helium atmosphere would virtually eliminate absorption in the gas and would also permit higher specific power for any specified limiting graphite temperature. These advantages would quite possibly justify the use of helium in the United States. Nevertheless, nitrogen has been assumed in this study in order to be consistent with the Russian design specifications.
Void Sensitivity

The fuel to graphite volume ratio of this reactor is such that the lattice multiplication factor is almost completely insensitive to the density (or void fraction) of the coolant. With the small water volume of the reactor, this property is achieved with little or no sacrifice in neutron economy.

Because of the insensitivity to water density no operating instability problems would be expected in connection with coolant boiling, changes in void fraction resulting from fluctuations in electrical load demand, or transition from water to steam cooling in the superheater tubes.

Temperature Coefficients of Reactivity

The prompt temperature coefficient of reactivity, composed of water and fuel temperature effects, is negative but becomes less negative with increasing fuel exposure. The graphite temperature coefficient is substantially negative at low exposures but becomes positive at about 1760 MWD, and becomes larger in magnitude than the negative metal-water coefficient at about 3300 MWD/T. At approximately 8800 MWD/T, the overall positive temperature coefficient effect is larger in magnitude than the equilibrium xenon poisoning.

Fuel Management

Basically three fuel management schemes have been considered for this reactor. These are:

a. Full pile discharge at 2200 MWD/T average.

b. Discharge at 5500 MWD/T average with annual replacement of spent fuel.
c. (1) First load - discharge at 5500 MWD/T average with annual replacement of spent fuel.

(2) Replacement loads - discharge at 11,000 MWD/T average with annual replacement of spent fuel.

All three are feasible operating methods; the choice between them will be based on economic or fuel performance considerations.

Reactor Control

The reactivity control system capacity required for operation of this reactor with unirradiated fuel must be adequate to compensate for the negative temperature coefficient, the buildup of fission product poisons, and some degree of fuel burnup. Approximately 90 mk would be required for operation on a full load discharge basis at 2200 MWD/T average exposure, the specified operating conditions of the Russian reactor.

For operation to higher exposures, higher initial enrichments are required and the total requirement for startup with an unirradiated fuel load is increased. Conversely, the control requirement for reactor startup and operation at average exposures of several thousand MWD/T is reduced because of the positive graphite temperature coefficient at higher exposures. The control system chosen for the modified design is adequate for routine operation at an 11,000 MWD/T discharge exposure with annual replacement of spent fuel except that the discharge exposure of the initial loading must be reduced to an average of 5500 MWD/T with the corresponding reduced enrichment necessary to permit startup of the unirradiated reactor. Alternatively, the control system will permit operation to 5500 MWD/T discharge exposure with annual fuel replacement and no restriction on the
initial loading, or operation on a full pile discharge basis to about 3300 MWD/T.

The functions of the reactivity control system are:

a. Compensation for startup and shutdown reactivity transients.

b. Compensation for long term reactivity transients associated with isotopic buildup and burnout.

c. Rapid shutdown of the reactor when conditions require, and

d. Control of the operating power level and distribution.

The total capacity of the system is determined by requirements (a) and (b) plus the capacity needed to fulfill requirement (d). The control system chosen consists of 241 rods with a capacity of approximately 110 mk which is adequate under the operating conditions described previously.

The requirement for rapid shutdown, or scram of the reactor may be broken down into the immediate and the ultimate requirements. The immediate requirement is that needed to override the prompt temperature coefficient effects, which for the unirradiated reactor amount to 18 mk. Fifty-five rods with a capacity of about 25 mk are provided with a "fail-safe" rapid insertion system for this function. The ultimate control requirement includes the additional effects of the delayed temperature coefficient (due to graphite temperature) and xenon decay. These constitute 35 mk in the unirradiated reactor. All rods not equipped for fast scram, except the 24 operating control rods, are required to enter the reactor at normal speed following a scram signal. The additional control capacity
thus provided will depend upon the number of the rods already inserted
during operation but will be at least the required amount.

For control of the operating power level and power distribution, 24 operat-
ing control rods are provided in locations permitting control of localized
flux or power perturbations as well as over-all power level control. These
rods can be manually controlled. Automatic control may be feasible as
well with provision of appropriate sensing and control instrumentation.

The above discussion and the following tabular data apply specifically
to the reactor with magnesium matrix fuel, and also generally for 9%
molydenum alloy fuel.
### TABLE I
PHYSICS PARAMETERS

I. Neutron Multiplication Cycle

Zero exposure fuel at operating conditions;
Enrichment = 1.3%; boiling lattice.

<table>
<thead>
<tr>
<th>Neutrons produced in fission</th>
<th>Thermal</th>
<th>Epithermal</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parasitic capture</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel (U-235 + n → U-236)</td>
<td>7.0</td>
<td>.8</td>
</tr>
<tr>
<td>Steel</td>
<td>7.1</td>
<td>1.4</td>
</tr>
<tr>
<td>Water</td>
<td>1.6</td>
<td>.6</td>
</tr>
<tr>
<td>Other</td>
<td>3.1</td>
<td>.8</td>
</tr>
<tr>
<td>Pu-producing capture</td>
<td>13.6</td>
<td>19.9</td>
</tr>
<tr>
<td>Leakage</td>
<td>.5</td>
<td>1.0</td>
</tr>
<tr>
<td>Control rod capture</td>
<td>1.5</td>
<td>.9</td>
</tr>
<tr>
<td>Fission-producing capture</td>
<td>34.9</td>
<td>5.3</td>
</tr>
<tr>
<td>(Neutrons produced)</td>
<td>(86.2)</td>
<td>(13.8)</td>
</tr>
</tbody>
</table>

II. Average Core Flux Levels

Specific power = 3.84 kw/kg
1.3% enrichment

<table>
<thead>
<tr>
<th></th>
<th>Thermal</th>
<th>Epithermal</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1.3x10^{13}</td>
<td>2.53x10^{13}</td>
</tr>
</tbody>
</table>

III. Short-Term Transient Reactivity Effects

Change in multiplication ($\Delta k$) in going from cold to operating conditions

<table>
<thead>
<tr>
<th></th>
<th>$\Delta k$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Exposure (MWD/T) (kw-D/kg)</td>
<td>0 4,000</td>
</tr>
<tr>
<td>Water and Fuel Temperature ($T_f = T_w$)</td>
<td>-.0166 -.0099</td>
</tr>
<tr>
<td>Additional Fuel Temperature</td>
<td>-.0011 -.0011</td>
</tr>
<tr>
<td>Graphite Temperature</td>
<td>-.0144 +.0175</td>
</tr>
<tr>
<td>Void Formation</td>
<td>~ 0 ~ 0</td>
</tr>
</tbody>
</table>
Doppler Coefficient \[ \frac{1}{k} \frac{\Delta k}{\Delta T_f} = -3.2 \times 10^{-5}/\degree C \]

Xenon Poisoning = -20 mk

Samarium Poisoning = - 6 mk

IV. Long-Term Reactivity Changes

\[ \frac{\Delta k_{\infty}}{k_{\infty,0}} \]

Initial Enrichment

<table>
<thead>
<tr>
<th>Fuel Exposure MWD/T</th>
<th>0</th>
<th>0</th>
<th>0</th>
<th>0</th>
</tr>
</thead>
<tbody>
<tr>
<td>2,200</td>
<td>-.0002</td>
<td>-.0087</td>
<td>-.0138</td>
<td></td>
</tr>
<tr>
<td>5,500</td>
<td>-.0469</td>
<td>-.0559</td>
<td>-.0603</td>
<td></td>
</tr>
<tr>
<td>11,000</td>
<td>-.1226</td>
<td>-.1308</td>
<td>-.1343</td>
<td></td>
</tr>
</tbody>
</table>

Initial Conversion Ratio

| .70 | .65 | .61 |

V. Control Capacity

Local Control Rod Strength

(-equivalent $\Delta k$ in a region of 8 square lattice units area)

Total System Strength (241 rods)

0.124

0.112
The radial power distribution would be between curves 1 and 2 depending on the degree of fuel burnout. The fraction of reactor power generated in the superheat section is:

<table>
<thead>
<tr>
<th>Curve</th>
<th>Superheat Power Fraction</th>
</tr>
</thead>
<tbody>
<tr>
<td>Curve 1</td>
<td>.291</td>
</tr>
<tr>
<td>Curve 2</td>
<td>.301</td>
</tr>
</tbody>
</table>
B. **HEAT TRANSFER AND FLUID FLOW**

The important thermal and hydraulic performance data are summarized in Table II.

1. **For Magnesium Matrix Fuel**

   **Boiler Region**

   The important thermal limitations are fuel element and cladding temperatures, surface heat flux and exit steam void fraction.

   The maximum fuel element temperature is 729 °F. Since the melting point of magnesium is 1200 °F, fuel element damage appears remote even during the most severe power excursions. The maximum cladding temperature is 670 °F which is well below the tolerable corrosion limits for stainless steel. The axial temperature distribution is shown in Figure 1.

   The importance of the surface heat flux arises from burnout considerations. When the heat generation is described by a cosine or "chopped" cosine, the critical location is downstream of the fuel mid-point. Therefore, the critical heat flux is not the maximum fuel element heat flux. For these conditions, the critical point is 15.6 feet from the inlet of the active zone. The surface heat flux is 169,500 B/hr-ft². The calculated(5) burnout heat flux is 373,000 B/hr-ft² at this location. The maximum surface flux is 192,000 B/hr-ft² and its corresponding burnout flux is 521,000 B/hr-ft². These latter values are given in Table II. However, it should be kept in mind that at the 15.6 foot position, the margin of safety is minimum. Further calculations show that either a 100 percent power increase or a 50 percent flow reduction with other conditions being equal will cause burnout.
It was assumed that the exit quality is constant in the core. The bulk outlet quality is then 36.4 percent for all fuel assemblies. By Armand’s Correlation\(^6\), the steam void fraction assuming slip flow is 74.5 percent. It has been reported that flow in parallel vertical tubes becomes hydraulically unstable when the exit void fraction exceeds 70 percent. This value, however, appears to be based on low pressure data. Greater stability has been realized at high pressures. The Russians indicate that stable flow was obtained with 50 percent exit quality at 1200 psig by means of an upstream orifice. The void fraction for these conditions is 79.4 percent. Thus, if it is assumed that stable flow is limited to 80 percent exit void fraction, a 13.5 percent power increase will be permissible. Similarly, a flow reduction of 12.0 percent will be the limiting value.

The hydraulic demand curve was calculated to determine the boiling stability of the system. The two-phase pressure drop calculation was performed using Armand’s Correlation\(^6\). The outlet connector which extends from the fuel assembly outlet to the steam separator was assumed equivalent to 804 diameters of 1.0 inch ID pipe. The inlet connector was assumed equivalent to 1235 diameters of 1.0 inch ID pipe. The pressure drop across the flow stabilizing orifice was assumed 15 psi. The demand curve which is the pressure drop from the inlet of the active zone to the inlet to the steam separator is shown as a function of the flow squared in Figure 2. The supply curve is the pressure drop from the distribution header to the inlet of the active zone. As seen a point of instability does not occur.
2. **Superheat Region**

The most important thermal limits are fuel element and cladding temperature.

The maximum fuel element temperature is 1099 F, which is 101 F less than the melting point of magnesium. Thus, before melting results either a 20 percent power increase or an 18 percent flow reduction with all other conditions remaining equal must first occur.

The maximum cladding temperature is 1045 F. The value is less than the corrosion limit. However, the fuel element melting point is less than the cladding corrosion limit, therefore, the cladding temperature is only of secondary importance under these operating conditions.

The normal axial temperature distribution is shown in Figure C.

3. **Conclusions**

The calculations performed indicate that the reactor will operate satisfactorily at the design conditions and with 10 percent changes in either the flow or power in both the boiler and superheat regions. The power level increase is limited to 13.5 percent in the boiler region by exit steam void fraction considerations.

Further power level increases will require a flow increase and/or greater subcooling at the inlet. Both modifications will decrease the exit void volume. Burnout limits indicate that a 100 percent power increase is possible.

The superheat fuel elements are limited to a 20 percent power level increase by the fuel element temperature. Greater power increases will require increased flow and/or lower inlet temperature.
<table>
<thead>
<tr>
<th></th>
<th>Boiler Region</th>
<th>Superheat Region</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>Coolant Water</td>
<td>Steam</td>
</tr>
<tr>
<td>2.</td>
<td>Outlet Temperature (F)</td>
<td>644</td>
</tr>
<tr>
<td>3.</td>
<td>Inlet Temperature (F)</td>
<td>572</td>
</tr>
<tr>
<td>4.</td>
<td>System Pressure (psig)</td>
<td>2,200</td>
</tr>
<tr>
<td>5.</td>
<td>Coolant Flow (lb/hr)</td>
<td>8.43x10⁶</td>
</tr>
<tr>
<td>6.</td>
<td>Average Coolant Flow Rate(lb/hr.)</td>
<td>5,440</td>
</tr>
<tr>
<td>7.</td>
<td>Inlet Core Velocity (ft/sec)</td>
<td>7.8</td>
</tr>
<tr>
<td>8.</td>
<td>Outlet Core Velocity (ft/sec)</td>
<td>25.3</td>
</tr>
<tr>
<td>9.</td>
<td>Slip Ratio (outlet)</td>
<td>1.3</td>
</tr>
<tr>
<td>10.</td>
<td>Gas Velocity - Liquid Velocity (ft/sec)</td>
<td>6.65</td>
</tr>
<tr>
<td>11.</td>
<td>Void Fraction (outlet)</td>
<td>.745</td>
</tr>
<tr>
<td>12.</td>
<td>Maximum Fuel Temperature (F)</td>
<td>729</td>
</tr>
<tr>
<td>13.</td>
<td>Maximum Cladding Temperature (F)</td>
<td>670</td>
</tr>
<tr>
<td>14.</td>
<td>Burnout Heat Flux (B/hr-ft²)</td>
<td>521,000</td>
</tr>
<tr>
<td>15.</td>
<td>Maximum Core Heat Flux (B/hr-ft²)</td>
<td>192,000</td>
</tr>
<tr>
<td>16.</td>
<td>Average Core Heat Flux (B/hr-ft²)</td>
<td>90,600</td>
</tr>
<tr>
<td>17.</td>
<td>Average Core Power Density (kw/t-ft³)</td>
<td>41.5</td>
</tr>
<tr>
<td>18.</td>
<td>Peak to Average Power Ratio (axial)</td>
<td>1.5</td>
</tr>
<tr>
<td>19.</td>
<td>Peak to Average Power Ratio (radial)</td>
<td>1.41</td>
</tr>
<tr>
<td>20.</td>
<td>Average Specific Power (kw/t/kg-U)</td>
<td>3.84</td>
</tr>
<tr>
<td>21.</td>
<td>Fuel Management</td>
<td>-</td>
</tr>
<tr>
<td>22.</td>
<td>Average Fuel Burnup (MWD/mt)</td>
<td>-</td>
</tr>
<tr>
<td>23.</td>
<td>Peak to Average Burnup Ratio</td>
<td>-</td>
</tr>
<tr>
<td>24.</td>
<td>Material Makeup Rate (lb/day)</td>
<td>-</td>
</tr>
</tbody>
</table>
AXIAL TEMPERATURE DISTRIBUTION IN MAXIMUM BOILER REGION FUEL ASSEMBLY

FIGURE C.
HYDRAULIC DEMAND CURVE FOR MAXIMUM BOILER REACTION FUEL ASSEMBLY

**Figure D:**

- Supply Curve
- Demand Curve

Axes:
- Y-axis: 0 to 70
- X-axis: (w/v₀)² from 0 to 1.0
AXIAL TEMPERATURE DISTRIBUTION IN MAXIMUM SUPERHEAT REGION FUEL ASSEMBLY

FIGURE E.
C. RUSSIAN SUPERHEAT REACTOR AS AFFECTED BY GRAPHITE PROPERTIES AND STACK DESIGN

Graphite Thermal Conductivity

The rate of heat dissipation, per unit volume of graphite, will be much less in the Russian reactor than in existing Hanford reactors. The Russian reactor graphite temperatures will be fairly independent of thermal conductivity for values of 10 Btu/hr-°F-ft and higher since the heat flux in the graphite is low. Graphite temperature will be very dependent upon the thermal conductivity of the fuel since all heat produced in the graphite must flow to the internally cooled fuel assembly.

Graphite Contraction

The graphite contraction rate in the Russian reactor will be much lower than in Hanford reactors since specific tube powers are lower. The graphite contraction rate in the Russian reactor should not exceed 0.02 percent per year or a total of 0.6 percent in a 30-year life. In the 3.4-foot diameter of the active zone, this would be 2.5 inches or 1.3 inches between the core center and fringe.

Information on detailed arrangement of the graphite blocks within the Russian reactor core is not available. Means of horizontal interlocking of the vertical columns of graphite must be provided for earthquake stability and steam venting (see below). A rigidly designed core could cause 1.3 inches of inward deflection of the fringe tube channels after 30 years of operation. This will require that flexibility be designed into the fuel assemblies and control rods to allow insertion into a bowed channel. It should be noted that 1.3 inches is a fraction of the bowing to which some NPR process tubes and control rods will be subjected.
Steam Venting

Provisions to vent steam from a ruptured coolant tube must be added to the reactor core. There is no indication of any consideration of this in the original Russian design, but it was mentioned in a recent Vienna paper. The graphite blocks can be cut out as shown in SK-1-4562* to provide vertical vent passages. Also, horizontal passages can be provided by undercutting the graphite blocks. The result will be a honeycomb of vertical and horizontal passages, similar to NPR, which will dissipate the steam pressure and energy. A duct will have to be run from the reactor gas container to the fuel discharge pool to act as a relief valve for the escaping steam.

It should be noted that the design of the Russian fuel assembly will result in lower rupture steam and water flow than in the case of an externally cooled fuel element such as in NPR. The actual steam vent areas required for the Russian reactor must be determined by two-phase flow calculations and mockup tests.

Stability Under Earthquake Forces

The graphite stack should be designed to resist earthquake forces due to accelerations of 0.2 g, which is greater than the maximum earthquake intensity of 8 on the Rossi-Forel scale that has been recorded for the hypothetical reactor site. Stack rigidity can be provided by using notched horizontal crosstie graphite bars to maintain vertical alignment of the tube blocks. Notching of the vertical tube block may also be necessary. The result will be a matrix of horizontal and vertical graphite bars assembled in a "Lincoln-log" arrangement, similar to the NPR design.

* Note: A small revision of SK-1-4562 is necessary.
Burnout

With the nitrogen gas atmosphere containing only trace amounts of oxygen and carbon dioxide, it would appear that except for burnout due to water or steam leaks into the core there should be negligible burnout of the graphite.
D. FUEL

1. Magnesium Matrix

Fabrication and Irradiation Experience at HAPO*

U-Mg fuel material has been fabricated at HAPO using uranium punchings, machine turnings, and U-metal shot with pure Mg, Mg-Si, and Mg-Zr alloy as the matrix. Uranium contents as high as 65 v/o U have been produced using shot, and as low as 30 v/o using machine turnings. Mg-Si alloy as a matrix material appears more desirable than pure Mg since a bond between the matrix and the fuel may be obtained. Mg-Zr alloy also has potential, but chloride salts present in the Mg-Zr master alloy have proven difficult to remove.

Initial irradiation testing at HAPO is reported in HW-43973, August 1, 1956 (Confidential--now declassified with deletions). In this irradiation, twelve small specimens were canned in Zr-2 capsules with an 0.050" wall. Examination after irradiation (up to 2.0 v/o burnup) revealed that the specimens were dimensionally stable at the test conditions. Based on pre and post irradiation bend testing, ductility decreased and strength increased slightly as a result of the irradiation. These effects appeared to have occurred by about 0.1 percent burnup with little additional changes observed at increased exposures.

Two larger U-Mg fuel material specimens were irradiated (MTR) to a goal of 10,000 MWD/T of contained U. These specimens were 64 v/o U and fabricated from Mg-Si alloy and U-shot. They were 1.00" diameter x 4.00" long and canned in 0.060" thick Zr-2 cans. Interim examinations at the MTR at about 2000 and 6000 MWD/T indicated no gross dimensional instabilities. The recent examination of one of these specimens

*Weights in Section D are in short tons.
at a calculated 9,900 MWD/T (burnup, analysis 5,500 MWD/T) revealed no change in its 4.0" length and a 0.007" increase in its 1.000" diameter. The fuel specimen shows a tendency to swell, but has been restrained by the heavy walled Zr-2 can.

An aluminum canned U-Mg specimen has been irradiated to about 800 MWD/T at the MTR. The specimen was fabricated from enriched U turnings, Mg-Si matrix, and was about 35 v/o U. Post-irradiation examination revealed a maximum growth of 0.025" on the 1.440" diameter and an increase of 0.009" on the 4" length of the canned specimen.

Powder metallurgy U-Mg specimens prepared by KAPL have also been tested up to 3000 MWD/T. These specimens were 0.375" diameter x 1.50" length and were encapsulated in 0.136" thick aluminum. Post-irradiation examination revealed fuel material swelling without much apparent restraint by the heavy aluminum cladding.

A resume of swelling data follows:

<table>
<thead>
<tr>
<th>Material</th>
<th>U Content</th>
<th>Clad</th>
<th>Exposure</th>
<th>Volume Percent Increase</th>
</tr>
</thead>
<tbody>
<tr>
<td>KAPL Powder Met 0.375&quot; x 1-1/2&quot; long</td>
<td>61.6 v/o</td>
<td>0.136&quot; Al</td>
<td>1000 MWD/T</td>
<td>2.63%</td>
</tr>
<tr>
<td></td>
<td>66.3</td>
<td>0.136&quot; Al</td>
<td>1000</td>
<td>4.41</td>
</tr>
<tr>
<td></td>
<td>64.3</td>
<td>&quot;</td>
<td>3000</td>
<td>9.74</td>
</tr>
<tr>
<td></td>
<td>61.8</td>
<td>&quot;</td>
<td>3000</td>
<td>10.2</td>
</tr>
<tr>
<td></td>
<td>67.1</td>
<td>&quot;</td>
<td>3000</td>
<td>8.56</td>
</tr>
<tr>
<td>HAPO-1-3/8&quot; x 3.6&quot; long</td>
<td>35</td>
<td>0.050&quot; Al</td>
<td>800</td>
<td>2.7</td>
</tr>
<tr>
<td>HAPO-1&quot; x 4&quot; long</td>
<td>64</td>
<td>0.060&quot; Zr-2</td>
<td>9900</td>
<td>1.4</td>
</tr>
<tr>
<td>HAPO-0.40&quot; x 1-1/2&quot; long (12 pieces)</td>
<td>50</td>
<td>0.050 Zr-2</td>
<td>1000-20,000</td>
<td>Small</td>
</tr>
</tbody>
</table>

All above irradiations in cold water (∼40°C)
The data reveal a tendency of the U-Mg fuel material to swell as a function of exposure. Cold Zr-2 clad offers much more restraint than does the Al clad.

From this experience, there is nothing to indicate that the Russian fuel would not perform exceptionally well under the irradiation conditions stated. Even at 20,000 MWD/T, dimensional stability should not be a serious problem.

**Corrosion Behavior of U-Mg**

A deterrent to the application of U-Mg fuel material in water-cooled reactors is its poor corrosion resistance. The use of Mg as a matrix material affords the least corrosion resistance of almost any possible matrix material. In contact with water at high temperatures, Mg corrodes as rapidly as uranium and the corrosion product (MgO) is quite voluminous. In HAPO fuel elements purposely defected with .025 holes through the cladding, failure occurred rapidly. The times from the first indication of slug swelling to complete flow blockage in test loops operating at about 220-240 C were on the order of about 2-3 minutes. These times were so short that in-reactor testing of U-Mg fuel elements at HAPO was virtually eliminated. The fact that the fuel cores were not bonded to any of the three clad materials undoubtedly aggravated their failure performance. A bonded element might perform in a much more favorable manner, but the bonding of Mg to other materials remains unsolved.

Eutectic Mg-Si and Mg-Zr alloy matrix materials have been tested, but without any significant improvement in aqueous corrosion resistance. It is believed that any improvements in the aqueous
corrosion resistance of Mg by alloying is insignificant compared to that improvement needed to reduce the failure behavior of U-Mg matrix fuel material.

Potential Fuel Fabrication Techniques

(a) U-Mg Matrix

Among the core fabrication methods which hold good promise, perhaps die casting, power metallurgy or vacuum casting techniques are the most interesting. With a minimum of development effort, it should be possible to produce cores of sufficient dimensional and surface quality that subsequent machining is hardly required. High density cores free of internal voids are expected from any of the above techniques providing reasonable care is taken during the cleaning and outgasing steps.

The size of uranium particles suspended in the magnesium matrix may be important from an irradiation induced swelling consideration. In general, the finer the uranium dispersion, the lower the uranium content. The irradiation swelling may also be less for finely dispersed uranium particles. Shot of various sizes can be easily made by pouring molten uranium on a spinning graphite disk, and screening the resultant product. The techniques for manufacturing powdered uranium metal are available if needed.

(b) Graphite

The machining of graphite sections for the Russian reactor fuel elements should not offer any problems. It is expected that
tolerances would be in the order of \( \pm 0.005 \) to \( \pm 0.010 \) with the graphite sliding over the fuel cores relatively easily. In this way, even with some distortion, the graphite sections should be easily removed from the spent fuel elements prior to chemical processing. Since the graphite nearest the fuel channels is periodically replaced, it is expected the overall graphite distortion in the reactor would be reduced.

It is estimated the graphite sections are about 3.25" OD and may be any convenient length; possibly they could be made to finished size by an extrusion process.

(c) **Stainless Steel Tubes**

Stainless steel tubes in the diameter range of interest and down to about 0.003" wall thickness are being produced in this country by Allegheny Ludlum and others on a limited basis with normal radiographic inspection; there is no reason this technology could not produce quality tubes to meet the needs of this fuel element. The Russians have indicated the stainless steel is a type 1X18H9T.

Joining of thin-walled stainless steel tubes can be accomplished by at least three methods. These include: (1) Mechanical such as Swage-Lock fittings, (2) Heliarc or electron beam welding, or (3) brazing. The choice will depend largely upon the application, but should offer no problems which cannot be overcome with a minimum of development effort.
(d) **End Fittings**

Published pictures of the Russian fuel element suggest it may be possible to disassemble the spent fuel container and reuse the "upper-head." This practice would also allow removal of the graphite portions of the element before chemical separation.

The active portion of the fuel element would be severed near the top layers of graphite by sawing or using mechanical disconnectors. Assuming insignificant fuel distortion, the graphite could then be slid off the fuel elements. The base end of the upper-head suggests it was designed to allow reuse. Since it is made of stainless steel, decontamination in acid is possible.

Fabrication of the "upper-head" appears to be a normal machining operation with no unusual problems.

(e) **Fuel Core Assembly**

The core assembly is visualized as a slip-fit operation. Since the inner tube is continuous from the lower head to the upper connection, fuel failures caused by water leaks through fuel element closure zones should be minimized. Marginal weld quality on the anti-fragment cladding is less important because it is not exposed to high pressure - high temperature water. The attachment of the pressure tubes to the lower head and upper fittings should be a simple operation by either brazing or heliarc welding.

(f) **Comments Regarding Fuel Performance**

From the experience with U-Mg fuel at HAPO, there is little
reason to suspect the Russian fuel element would not perform well at the conditions stated. Even at exposures in the range of 20,000 MWD/T, little dimensional instability was observed. From this, it would appear the Russian reactor could operate at higher exposure levels with little danger.

The problem of U-Mg corrosion in a high temperature water reactor is serious under any condition. The Russian design, however, minimizes the possibility of a water leak by the absence of weld closures which are exposed to high temperature - high pressure water. The use of stainless steel also eliminates corrosion problems associated with jacket and closure failures. The design also takes advantage of the desirable properties of the stainless steel while minimizing the adverse effects.

The thermal conductivity of U-Mg lies between that of magnesium and uranium. For a 40 percent U-Mg slug, this is about

\[
\frac{.25 \text{ cal. cm}}{\text{sec. cm}^2 \text{ oC}} \text{ compared to .38 for Mg and .06 for U. This means the uranium will run cooler for any specific power generation than with pure uranium, while the swelling and dimensional distortion will be proportionally less. Since the pressure tube is not bonded to the core, some of this advantage may be lost. This is probably minimized by the pressure inside the tube which tends to hold the tube wall in intimate contact with the fuel core.} \]
PROCESS FLOW SHEET
(Assembly of Russian Fuel Elements)

FIGURE F.

Magnesium Alloy

- Prep of 0.4% MgSi
  - Cast Matrix
    - Core Finishing

Uranium

- Prep of U Metal
  - Molten Metal on Spinning Disk
    - Uranium Shot
      - Size Separation
        - Cleaning and Outgassing of Shot
          - Final Assembly
            - Inspection and Storage

Graphite

- Prep of Reactor Graphite
  - Machine Sections
    - Testing and Inspection
      - Graphite Expansion Form
        - Fabrication of Linear Expansion Absorber
          - Lower Head Unit
            - Assembly of Lower Head Unit & Expansion Absorber
              - Assembly of Cores on S.S. Tubes
                - Weld Anti-Fragment Cladding to Inlet Tube
                  - Assemble Graphite Sections
                    - Final Assembly
                      - Inspection and Storage

S.S. Tubes

- Prep of Thin Wall S.S. Tubes
  - Cleaning
    - Testing and Inspection
      - Machining of Parts

End Fittings

- Casting of End Fitting Parts
FIGURE G.

Fuel Cores in Place on Pressure Tubes

Anti-Fragment Shield 0.008" Can S.S.

S.S. Tubes 0.354 ID x 0.015 Wall

Shield

Diffusion Plate

Weld

Welded Connections

Graphite Rings

Upper Head

Fuel Section

Weld

Lower Head

ASSEMBLY OF FUEL ELEMENT
Upper Head

Shielding

Work Space for Making Connections

Access Cover

Graphite

Fitting

Fuel

FIGURE II.
.015" - .016" S.S.

.008" S.S.
Anti-Fragment Cladding

U-Mg - 34.5# of \( \begin{cases} \text{12 Wt.\% Mg-88\% U or} \\ \text{1.3\% Enriched U (U-Mg = 40\% U by Vol.)} \end{cases} \)

.906 OD x .354 ID x 19.7' Long (Approx. Dim)

Total U-Mg Wt. = \( \sim 39.2 \) 

One possible method of assuring the inside pressure tube is capable of expanding under pressure to give maximum heat transfer.

FIGURE I.
Disassembly Procedure

Because of cost considerations, it appears desirable to reuse the "upper-head" from charge to charge. It is believed this could be done by cutting the fuel element just above the upper-most graphite ring as shown on Figure H. This work would be done remotely.

The graphite would then be removed from the fuel cores either by shaking, or could be crushed if sticking is encountered.

The stainless steel "upper-head" with the stubs of pressure tubing still attached could then be decontaminated by an acid dip in a low level radiation facility. The braze joints could then be "unsoldered" and prepared for the next fuel assembly.

2. U-9 Percent Alloy Fuel

This section concerns the use of a 9 weight percent molybdenum-uranium alloy as the fuel material for the Russian steam-cooled reactor. As noted in the draft of the original report, the Russian sources of information do not specifically state that this type of fuel will be used in the reactor, but it can be inferred that it may be so used from statements made in Reference (4), (page 4).

It should be noted that Hanford has less first-hand experience with this fuel material than it has with magnesium matrix, and any statements made about the former come primarily from other sources.

A basic assumption is that the amount of uranium per fuel element in the two alternative types of fuel is equal.
Uranium-molybdenum alloy has been of interest as a power reactor fuel material since the early 1950's when WAPD begin development work on the CVR. When the CVR project was dropped, the work was applied to the PWR project. The poor corrosion resistance of uranium in hot water has been the incentive for alloy screening studies since 1943. In about 1952, WAPD began an intensive alloy development effort in support of the Naval Reactor Program. Alloying additions which had shown some merit in the early screening studies were subjected to further development and testing. A large part of our knowledge of uranium alloys was produced in this effort. By early 1954, the U-Mo alloy system was selected as the reference fuel material on the basis of the corrosion performance of U-12 Mo. Later, it was decided that the PWR should be fueled with UO₂. U-12 Mo was not used because they were not able to get reproducibility in quality between different melts. This occurred in the early days of arc-melting and would not be a problem today. Also working against U-12 Mo was the discovery, in work with the critical assembly, that resonance absorption of neutrons by Mo was far greater than anticipated.

In order to establish perspective of the critical criteria for the fuel material for the choice of fuel for the PWR, it must be re-examined. The requirement of greatest impact pertaining to the aqueous corrosion resistance of the fuel material. The fuel rods were required to behave such that the reactor could complete 125 days of continuous operation at full power, even though the fuel is partially exposed to water during this period because of some defect in the cladding. This is a very stringent requirement. Further, it was required that the fuel be dimensionally stable under irradiation, and that it be thermally
stable (i.e., undergo no transformations that would adversely alter its properties) during the irradiation period. Maximum fuel temperatures up to about 450 C and fuel surface temperatures to 370 C were anticipated. Although comprehensive summary reports on the alloy development program are available, a quantitative statement of what constituted dimensional stability is not available.

In the last few years, APDA has elected to use the U-10 Mo fuel for the central breeder region of the Enrico Fermi reactor. Although this fuel is specified for the first load and perhaps the second, the irradiation performance demonstrated in the development effort to date indicates that this fuel is not adequate because of excessive swelling rates.

The British have reported limited test results on the swelling behavior of U-Mo alloys. These tests are part of a large alloy screening program for swelling resistance. Extensive development work has not been reported and no plans to use this fuel are known. The tests reported showed remarkable swelling resistance by U-10 Mo at a few specific high fuel temperatures.

Physical Metallurgy of U-Mo Alloys

The corrosion and irradiation stability shown by WAPD to be best were the alloys in the vicinity of 10% Mo. Since the cross section of Mo is not particularly low (2.5 barns) the interest is in using as little Mo as one can and still achieve the performance desired. Most alloy studies have been in the range 0-16% Mo. As an aid to understanding the metallurgical behavior of these alloys, let us examine the phase diagram, Figure J, as given by Rough and Bauer (BMI-1300). The maximum solubility of Mo in alpha uranium is 1.7% and in beta uranium is 0.4%.
In gamma uranium it is about 25% at 1280 C. The alpha, beta and gamma phases have the usual structures of uranium, modified slightly in lattice parameter by the dissolved Mo. The epsilon phase has been shown to be an ordered form of the gamma phase. The delta phase is Mo.

The gamma phase decomposes by a eutectoid reaction, at 562 C and 11.4% Mo, to alpha and epsilon. The decomposition reaction has two aspects - precipitation of alpha and ordering of the residual gamma phase.

Quenching from the gamma field will retain the gamma phase. Much of the irradiation behavior and the corrosion behavior depend on whether these alloys are used as quenched gamma, or whether the transformation has been permitted to start by annealing, or whether transformation starts during service because of the time and temperature conditions. The transformation kinetics are an important aspect of the metallurgy of these alloys. The TTT diagrams are given in Figure K, where only the beginning of transformation is shown. It has been difficult to define the completion of transformation.

The kinetics have been shown to be altered by irradiation so that with the flux above a certain level, for a given temperature, the transformation does not occur. Interpretation of the irradiation behavior is very complex for this reason.

The corrosion behavior of these alloys in high temperature water is characterized by a low uniform penetration rate which varies with temperature in the usual way. However, a significant fraction of the corrosion-produced hydrogen is picked up by the metal and precipitates throughout it as platelets of hydride. After a rather long period of
exposure to water, the metal cracks into small pieces. This is called "discontinuous failure." It has been found that for U-12 Mo, 35 days annealing at 400 °C gave optimum corrosion life.

For PWR application, the improved corrosion resistance was important. For reactors with more freedom to shut down when a failure is detected, the improvement is of little consequence. Typical results are, at 340 °C, for U-10.7 Mo

quenched - 0.23 mg/cm²-hr, failure 28-42 days

quenched - annealed - 0.13 mg/cm²-hr, failure in 20 days.

Defected, clad specimens behave much better, presumably because of the protection from the hydrogen pickup offered by the cladding. Zircaloy-2 clad U-12 Mo fuel rods with a 40 mil hole through the cladding have withstood 275 days in 343 °C water without discontinuous failure occurring.

Some properties of these alloys are given below:

**Thermal Conductivity** (9-15% Mo)(10-100 °C)

0.34 cal/sec cm²-°C/cm

**Thermal Expansion** (gamma quenched U-10 Mo)(100-400 °C)

13.5 x 10⁻⁶ - in / in °C

**Melting Point** (U-12 Mo)

1150 °C

**Density** (gamma quenched U-10 Mo)

17.25 gm/cm³

**Recrystallization**

For 75% cold work, 1 hour at 600 °C gives a recrystallized structure.

**Tensile Strength**

100,000 psi at 370 °C

25% reduction of area

5% elongation
Irradiation Behavior

As mentioned in the preceding section, the behavior of U-10 Mo alloys depends on whether it is used in the gamma-quenched or the annealed condition. In the former state, it exists as a metastable, cubic solid solution which will have a given set of properties which will change if the time and temperature conditions will permit the onset of transformation. If the material is being used in an annealed condition, then the changes in properties will have already occurred and must be recognized.

In discussing irradiation effects, the state of the alloy is very important for several reasons. First, the gamma phase, being cubic, is isotropic and does not undergo growth while the precipitated alpha phase in the annealed alloy will grow. The epsilon phase, because of its crystal structure, would not be expected to grow either, although this does not appear to have been demonstrated. The problem that has caused the greatest concern is swelling and the swelling behavior seems to depend on the extent to which transformation has occurred or is occurring. It is reasonable that the movement of phase boundaries through the metal during transformation could influence the nucleation and growth of fission gas bubbles. A further effect that has appeared and is under study is the stabilization of the gamma phase by the fission events, providing that the fission rate is high enough. That is, for each fuel temperature, there is a fission density value above which transformation will not occur. From what is known now, practical reactors could not avoid having a substantial portion of the fuel below this critical level.

WAPD corrosion testing data after irradiation (up to 2000 MWD/T) showed no deterioration of corrosion resistance - if anything, there was an improvement. Post-irradiation bend tests of bare fuel pieces irradiated
to 1000 MWD/T showed that essentially no ductility remained. There are no data on thermal conductivity changes during irradiation.

The early irradiations of gamma quenched alloys showed that if the fuel temperature was somewhat below 400 C or lower, the fuel was quite stable - little or no growth was detected and the volume increase amounted only to 3-4% at 28,000 MWD/T. In several irradiations performed in the gamma phase, similar results were obtained. The temperature domain between 400 and 600 C was relatively unexplored although a few tests performed slightly above 400 C had suggested a threshold. From the point of view of dimensional stability, the U-Mo system looked good in 1955.

Later work with these alloys and others with similar metallurgical characteristics clearly demonstrated the importance of the transformation kinetics on swelling. When these materials are irradiated under time-temperature conditions such that transformation can occur, the swelling rates are excessive. The "nose" of the transformation curve for U-10 Mo is at about 450 C while that for U-10 Mo-4 Zr is at 550 C. The latter alloy behaves very well at temperatures up to 400 or slightly above out to burnups of 3 to 4%. At 425 C gross swelling has occurred by 0.5% burnup.

APDA studies with prototypic fuel elements in test reactors have shown the same effects except that the additional effect of fission rate has been discovered. If the fission rate is high enough, the metal will all transform to the gamma phase and will behave very well. Because they are using this fuel in a fast reactor with a power density several orders of magnitude greater than achieved in thermal reactors, this
effect may help them out, although it is probably not of use to those concerned with thermal types. Current estimates are that their fuel will be good to 1% burnup. They plan periodic discharge as the exposure increases to monitor fuel performance and determine the fuel limit this way. This fuel is regarded as of marginal value to them.

In summary, the U-10 Mo fuel may be a very useful type if the user can keep the fuel temperatures below 400 C or above 600 C. In between, there are serious limitations on the achievable exposure levels.

Fabrication

In general, uranium alloys of these compositions are fabricated in much the same way as unalloyed uranium. The following comments regarding the fabrication techniques of U-Mo alloys are based on experimental billets of up to 2 inches in diameter. There is reason to believe these results could be extrapolated to billet sizes required for production use. Figure 1 shows a flow diagram which indicates some of the possible routes for producing U-Mo fuel rods which have been evaluated.

(a) **Billet Fabrication**

**Casting**

Using zirconia washed graphite crucibles, the alloying of uranium with molybdenum has been accomplished by the direct addition of elemental Mo during the vacuum induction melting operation. Consumable arc remelting has also been attempted with no unusual difficulties noted.
Forging

Castings have been forged into octagons with 15:1 reduction in area with repeated heating in chloride salt at 1950 F. Some grain refinement has been reported but the structure is not uniform. With U-12 w/o Mo, some cracking was observed in the billets.

Extrusion

Cast U-1 2 w/o Mo has been successfully reduced in size by heating in salt at 1950 F for 12 minutes and extruding at a 10:1 reduction ratio. The grain structure was found to be uniform in both the transverse and longitudinal directions. Lower pressure is required to extrude a billet of wrought material than one of cast uranium.

(b) Extrusion Cladding Fuel Rods

Following a machining operation, cast or wrought billets have been sealed in Zircaloy-2 components, heated to about 1520 F for 12 minutes, and extruded at a 10.6:1 reduction ratio. The grain size of the extruded rod is reported to be considerably finer than in the starting material, regardless of the billet fabrication history.

No significant variation in grain size has been noted along the length of experimental rods produced by this process.

Surface finishes between 30-40 micro inches have been produced on clad rods using several combinations of time, temperature and extrusion ratios. A molybdenum disulfide and Led-Plate extrusion lubricant was found most effective in reducing surface galling, although glass, copper and steel have also been used successfully.
Cladding thicknesses in inches of about .0038 with a maximum of .0059 and a minimum of .0025 have been achieved on experimental rods. Destructive examination of these pieces showed excellent bonding between the Zircaloy-2 and uranium alloys.

Research and Development

The total amount of funds required for research and development for either the 1967 or the 1975 plant is expected to be approximately the same for the molybdenum alloy and magnesium matrix fuels. The funds required, and the results expected to be achieved toward reduction of unit costs of energy, are expected to be approximately the same in the two cases. At this point, it cannot be ascertained whether or not there is greater potential in the alloy material.

The following tabulation indicates the somewhat different direction that the research and development would take in the two different cases:

I. Core Fabrication

<table>
<thead>
<tr>
<th>U-MAG</th>
<th>U-9% MO</th>
</tr>
</thead>
<tbody>
<tr>
<td>a. Minor effort to determine most optimum methods of fabricating matrix.</td>
<td>a. Material is tougher to work than natural uranium. May require additional effort to develop best process.</td>
</tr>
<tr>
<td>b. Development of diffusion</td>
<td>b. Same as in &quot;a.&quot;</td>
</tr>
</tbody>
</table>

II. Assembly and Disassembly Techniques

<table>
<thead>
<tr>
<th>U-MAG</th>
<th>U-9% MO</th>
</tr>
</thead>
<tbody>
<tr>
<td>a. Minor effort to determine best methods of assembly, brazing of stainless, dimensional tolerances, etc.</td>
<td>a. Fuel element weight about 2-1/2 times that of U-Mag. This may offer unusual assembly problem not found in U-Mag case.</td>
</tr>
<tr>
<td>b. Nondestructive testing procedure for components and finished fuel elements must be developed.</td>
<td>b. Same as for U-Mag.</td>
</tr>
<tr>
<td>c. Same as with U-Mag.</td>
<td></td>
</tr>
</tbody>
</table>
c. Testing of disassembly procedures, elimination of graphite, other wastes, and the decontamination of the "head end."

d. Does fuel need to be bonded because of heat transfer and uranium corrosion rate after penetration?

III. Materials

U-MAG

a. Fabrication techniques for thin walled stainless steel tubes may require some modest R&D in order to meet the physical and metallurgical conditions required for strength and corrosion.

U-9% MO

a. Same as for U-Mag.

IV. Irradiation Behavior

U-MAG

a. Operating temperatures.

b. Speed of reaction when U-Mag allowed to contact coolant.

U-9% MO

a. Determine core stability above beta transformation point.

b. Irradiation experience at higher exposures. This includes a study of mechanical, physical, and chemical properties for the materials under extreme irradiation conditions.

c. Failure mechanism of unbonded U.
3. **Fuel Cladding: 321 Versus 304**

Because of lack of information about the type of stainless steel used by the Russians in the fuel elements, it was assumed that type 304 was used. Later, it was learned that the Russian type Д06НМ more closely approximates AISI 321. The differences between these two types are discussed below.

The characteristics of type 321 versus type 304 stainless steel that are pertinent to the analysis of the Beloyarsk reactor evaluation are (1) resistance to corrosion by heated and superheated steam, (2) strength, and (3) cost.

The uniform corrosion resistance of 321 is somewhat less than 304; but 321 has immunity to inter-granular corrosion by virtue of its titanium content. In heated and superheated steam, carbide precipitation has little significance because water and/or steam do not attack them.

The titanium addition causes a significant quantity of relatively large inclusion, i.e., 321 is a "dirty" stainless steel. These inclusions reduce the effective metal cross-section. When exposed to corrosive media, they are relatively easily removed, thereby producing a pocket where crevice corrosion and/or concentration cell corrosion may develop.

In the balance, 304 probably outweighs 321 with respect to corrosion resistance in high temperature steam largely because of the "dirtiness" of 321.
Type 321 is stronger than 304 in terms of ASME allowable stress for temperatures greater than 100°F. At 670°F (maximum clad temperature in boiling region) the 321 allowable is 14,800 psi and the 304 allowable is 11,050 psi. Similarly at 1025°F (maximum clad temperature in superheating region), the 321 allowable is 13,500 psi and the 304 allowable is 8,800 psi. The reason for this difference is that 321 has superior creep and creep-rupture strengths. However, at the original design fuel tube wall thickness, the calculated hoop stresses (assuming no contribution from the fuel or the outer clad) are of the same order of magnitude as the short time yield strengths of the stronger "18-8" stainless steels (347, 316 and 321). A similar assumption and calculation on the outer cladding shows stresses in excess of their tensile strengths.

The Russian alloy is reported to have 1 percent titanium whereas an AISI 321 will have no more than 0.5 percent titanium. With 1 percent of titanium, it is possible that the alloy would be age hardenable, thereby producing strengths in excess of AISI 321.

In general, from strength considerations, type 321 is the better of the two alloys.

The cost of 321 tubing is of the order of 1.15 times greater than the cost of 304 tubing. The effect of this upon the estimated fuel fabrication costs is 4.55 to 4.9 percent increase.

This incremental cost will be reflected in higher unit energy costs of approximately 1 percent at 2200 MWD/T and 5500 MWD/T and 0.5 percent at 11,000 MWD/T.
U-MO EQUILIBRIUM DIAGRAM

FIGURE J.
TIME - TEMPERATURE - TRANSFORMATION DIAGRAM FOR U-MO ALLOYS

FIGURE K.
FLOW SHEET FOR FUEL ALLOY PROCESSING IN ROD FABRICATION

FIGURE L.

Induction Melt

Duplex Melting

Consumable Arc Melt

Forge → Extrude

Machine Components

Assemble with other Components

Extrude

Draw & Swage

Draw

Swage

Cut to Length
E. REACTOR CHARACTERISTICS RELATING TO USE OF PLUTONIUM FUEL

Calculations of the apparent plutonium value in the Beloyarsk Russian Superheat Reactor have concentrated on one method of fueling the reactor. The mode approaches equilibrium recycle by successively charging the reactor with the plutonium available from the previous cycle and the balance of reactivity is made up by cascade enriched uranium. By using successive irradiation steps, the value of the plutonium ashes as a reactor fuel can be taken into account with accuracy, but only for the resulting plutonium compositions. The system converges because with enough steps, the value of the last batch of ashes can be assigned zero or substantial values without altering the value of ashes from the first step. The analysis methods do not require this many steps although this situation was achieved in some instances to prove this point.

Plutonium fueling with batch irradiation in eight recycle series was studied. Five different $k_{\infty}$ values were used and, in addition, extra cases were rerun to evaluate the impact of allowances for different moderator temperature coefficients. Batch irradiation was used to conform with intent of the designers. Checks are being made with graded discharge as plutonium values and fuel costs are usually lower when evaluated in graded rather than batch modes, and the probability of operators converting to graded cycles is great. Generally speaking, the higher the initial enrichment value of the fuel in a batch cycle, the larger the fuel exposure and the lower the fuel costs. However, variations in reactivity during a batch irradiation must be handled by control rods and/or burnable poisons, the cost of which is difficult
to determine. Rigorously, these costs should be included as part of the fuel cost for each value of initial enrichment. To avoid this, fuel cost and plutonium value analyses were performed over a range of initial enrichment values and it was possible to select one which appeared plausible, all things considered.

By plotting the fuel costs for the uranium base cases, a practical minimum cost was determined at initial $k_\infty$ of 1.25. Therefore, this recycle series should probably be used as a standard for the batch cycles when examining the effect of the economic parameters on plutonium values in this reactor. Coincidently, the calculated plutonium values in this series are of the order of $14 \text{ per gram fissile nitrate}$, which has been found to be the mean value in a similar analysis in a water-moderated reactor (APWR).

The plutonium value analysis employed was made from a fuel cost relationship developed with HLO fuel burnup code, MELEAGER, and a fuel cost computation, QUICK. The essence of the value analysis, PUVE, is the development of a set of equations involving the unknown batch value of the plutonium fueling a reactor, the unknown batch value of the plutonium ashes leaving the reactor, and the fuel cost. The plutonium feed and ash values are generally different. The value of a batch leaving the reactor is conditionally determined by using that batch as feed for the next cycle, and so on. This is repeated until some logical constraint can be applied to the value of the last batch of ashes produced, thus removing the necessity of further cycles.

While lower initial $k_\infty$ gave higher costs, detailed examination was also made for an initial $k_\infty$ of 1.065, which is representative of the actual
Russian design. Under these conditions plutonium can have a higher value if one allocates to the productivity of plutonium the smaller change in reactivity from cold to hot moderator associated with plutonium as compared to uranium-235 enrichment. If one does this plutonium values of $30 to $40 per gram are calculated. It appears precarious, however, to allocate to plutonium this value, in view of the fact that the use of burnable poisons permit such an effective increase in initial $k_{\infty}$ that, in effect, the change due to different temperature coefficients is negligible. If burnable poisons are not permissible, and the control system limits the initial $k_{\infty}$ to 1.065, then the validity of the temperature coefficient argument to the value of plutonium is extremely important. Without a temperature coefficient bonus at low values of initial $k_{\infty}$, plutonium values are negative by as much as $150 per gram. The reason for this is that the initial fuel exposures are limited to about 2000 MWD/T, which produces plutonium very high in Pu-239. This plutonium, when inserted into the reactor, brings the reactivity to the control rod limit at very low fissile densities. These densities of plutonium, in turn, are not sufficient to support the economic fuel exposure.

The computer programs used for this computation have been arranged to bring out a complete report which is reproduced directly from the computing machine (HW-68867). The report relates the plutonium value as a function of the many parameterized variables in a manner such that they can be examined by scanning the summary sheets. In this form, the data can be used to facilitate the optimization of the over-all design with regard to total power costs and fuel cycles. Preliminary examination of graded, rather than batch, irradiation of fuels in this Russian machine appears to reduce fuel costs by roughly a factor of two.
Incremental fabricating and jacketing charges of plutonium fuel over uranium fuel have been treated in some detail. It is concluded that the impact upon plutonium value is more philosophical than practical if the incremental costs are less than $20/pound uranium jacketed. By zoning the reactor and confining plutonium enrichment to depleted uranium, 10 to 20 grams of fissile plutonium is used in each pound of fuel which would spread an incremental fabrication charge of $20/pound of fuel so that the plutonium value is reduced only one to two dollars per gram. If, on the other hand, the recycled plutonium is used to enrich every fuel element in the reactor, then the recycled plutonium is in two to three gram quantities in each pound of fuel and the associated plutonium value reduction is 10 to 20 dollars per gram for an incremental charge of $20/pound uranium. Zoning recycled plutonium has a slight cost because of the nonlinearity of the uranium price schedule and if the incremental fuel fabrication and jacketing cost were zero, zoning would be undesirable on this basis alone. Zoning of plutonium has virtues in its own right in that Pu-242 buildup in the recycled plutonium can be minimized by selling or even discarding the highly burned material. It appears that only two zones are necessary to secure the major benefit of this approach. Pu-242 is a small problem in this reactor, however, because of the extremely low specific power (3 MW/Ton) it takes 50 years of recycling plutonium to develop an even mildly deleterious amount.

Additional information on this portion of the study is available in HW-68479, "Plutonium Value Analyses for the Superheat Reactor, at Beloyarsk, Russia," undated.
F. SUPERHEATER PROBLEMS

Steam Separation and Solids Carry-Over

Dry steam is required at the entrance to the superheater section to (a) reduce the solids carry-over into superheater; (b) minimize erosion and radioactivity carry-over into the turbine; and (c) provide accurate thermal control of the superheater section.* In this design, the external evaporator removes the requirement for extreme separator compactness and further permits simple control of steam separation. The arts of water conditioning and external separator design have been developed to the point where carry-over of solids with the steam can be reduced to negligible amounts.

In general for a given boiler, the concentration of solids in the steam is dependent on the concentration in the boiler water, when other factors are equal. The water treatment intended for the boiler feed will make it possible to maintain total dissolved solids in the boiler water in the neighborhood of 10 PPM unless there is leakage of raw water into the system from the condenser. This leakage of raw water will be the determining factor on solids in the boiler and thus on solids in the steam.

Assuming no leakage from the condenser and proper operation of the water treatment equipment and boiler, it should be possible to maintain less than .01 PPM solids in the steam. This amount should not cause any significant problem in the superheaters or turbines.

*The problem of thermal shock of heat transfer surfaces by moisture carry-over is essentially removed since the vapor entering the superheater element will be throttled and slightly heated prior to contact.
Radioactivity Carry-Over to Turbine

Radioactivity can be transported to the turbine from two sources during normal operation. The first source is in-leakage of radioactive primary water into the boiler and transport by the steam to the turbine. This source of radioactivity should be very small because large leaks cannot be tolerated due to considerations such as waste disposal, etc. Also the concentration of radioactive solids in the steam will be no more than one percent and probably less than 0.1 percent of the concentration in the boiler water.

The second source of radioactivity is from ruptured superheater fuel elements. Work by Straub\(^{(7)}\) of the University of Illinois shows that many solids are soluble in superheated steam. It is likely that some of the rupture products will dissolve in the steam and deposit in the turbine. The magnitude of the problem caused by this mode of radioactivity transport is not known. It is significant that G. E. personnel have some evidence that this type of transport may be a problem and are investigating it in connection with their study at Vallecitos on the Superheat Advanced Demonstration Loop.\(^{(8)}\)

When the superheater is filled with water during shutdown and startup, radioactive material may be transferred from the boiler to the superheater. The amount should be small. The non-radioactive solids in the boiler water will be deposited in the superheater section when the water evaporates. The solids will be activated and some of them will dissolve in the superheated steam and deposit in the turbine. This need not be a serious problem if the boiler water is maintained at essentially demineralized water quality during shutdown and startup.
Corrosion

Corrosion or erosion of the stainless steel portion of the superheater should not be a problem. However, there is a possibility of serious corrosion of the carbon steel portions of the superheater loop where oxygen concentration in the loop may approach $1/4$ cc per liter. This amount of oxygen may cause serious corrosion in the boiler and piping.

Investigations to determine the seriousness of corrosion with oxygen laden water are being conducted by others, $(9, 10)$ and the results should be available within two or three years.

If the problem proves to be serious, a deaerator could be installed on the outlet of the reactor superheater tubes, or the entire system can be made of stainless steel.
VI. COMMENTS ON OPERATION

A. FUEL MANAGEMENT

Fuel replacement at the 2200 MWD/T exposure level is assumed to take place at approximately two-year intervals. The entire core load will be replaced. At 5500 MWD/T, it is assumed that approximately 25 percent of the core load will be replaced each year. At 11,000 MWD/T, it is assumed that approximately 12.5 percent of the core load will be replaced each year.

In approaching the 11,000 MWD/T average fuel exposure at discharge, it will be necessary to charge the reactor with 1.4 percent U-235 fuel and gradually replace the first batch, after an exposure of approximately 5500 MWD/T, with fuel of 1.7 percent enrichment which will be irradiated to 11,000 MWD/T. Fuel costs have been computed by averaging the costs for the first batch of material of the lower enrichment and later batches of material of the higher enrichment.

No fuel reshuffling within the various zones of the reactor is contemplated since the flux in the reactor can be flat enough to obtain adequate average exposure to realize most of the potential value of the fuel, particularly at the higher exposure levels which are considered to be feasible for this reactor.

As noted elsewhere in this report, it will be necessary to disassemble the fuel elements in the reactor plant and ship the fuel cylinders to the separations plant without the graphite and the end fittings. The top end fittings will be retained, decontaminated, and reused four times.
The costs allow for disposal of the parts of the fuel elements which cannot be reused.

B. UNIQUE OPERATION AND MAINTENANCE PROBLEMS

A study of the available literature and outline drawings of the subject reactor indicates the following operation and maintenance problems:

1. The connections and valving under the operating floor will be difficult of access for repair and test. Special methods and techniques will have to be developed to expedite the work.

2. The assembly and disassembly of the fuel elements will present new problems. Production-line assembly methods will probably be required for economic reasons.

   The disassembly of the elements will require use of highly-developed remote methods to minimize exposure and protect the components to be reused.

3. It is assumed that no "hot" shop will be provided. This will necessitate the analyzing of the economics of repair versus disposition of items of contaminated equipment which require machine repair such as metallizing and machining of pump shafts. For the disposal of equipment or parts it may be necessary to encase them to prevent spread of contamination.

4. The use of outside machine shops will require considerable follow-up and detailed engineering work at the plant.

5. Personnel will have to be trained thoroughly in the techniques of nitrogen handling. However, it is anticipated that the problems will not be as great as those experienced with helium systems.
C. PLANT AvAILABILITY ANALYSIS

Plant availability of 90 percent can be realized, based on the following assumptions:

1. Plant control instrumentation will be designed with extremely high reliability.

2. Equipment layout, piping, valving, etc., will be such that items can be isolated during operation.

3. Structural and equipment design and layout will be such that heat exchangers, pressurizers, etc., can be changed out and tube bundles replaced. This does not imply replacement during operation.

4. Outages will occur on an approximate quarterly basis, with two outages per year being extended for major equipment overhaul.

5. Rupture will occur in only 0.3 percent of the fuel tubes charged, or at 11,000 MWD/T, less than two per year. Some scheduled maintenance work will be performed during these outages.

6. Provisions will be made for automatic switchover to a second electrical source in case of a power failure in the primary electric supply system to essential plant auxiliaries.

7. The discharge operation will not interfere with outage maintenance. Five hundred of the 870 outage hours will be utilized for charge-discharge, sufficient time for refueling at 5500 MWD/T or higher exposures.

8. Outages can be scheduled with comparative freedom.

9. The training of both station personnel and short-term supplemental help is maintained at a level such that work can be accomplished expeditiously.

10. Radiation exposure will not control crew size or limit rate of work accomplishment.
VII. RESEARCH AND DEVELOPMENT PROGRAM - 1967 PLANT

In general the research required to build this plant will have been accomplished through reactor programs currently being sponsored by the Division of Reactor Development or the Production Division. Certain development programs would be required, however, to establish core design parameters to minimize design contingencies, and to assure reactor operability. The principal areas of nuclear design development, together with estimated minimum costs, are outlined below.

Physics

As in any large reactor program, the core physics analysis would require experimental confirmation. The Hanford experience with graphite-water lattice characteristics is sufficiently extensive that a critical assembly of this lattice configuration would not be necessary. However, confirmation of lattice parameters and control requirements are indicated, including the following: Measurement of $K_{oo}$, $\epsilon$, $p$, $\xi$; evaluation of Doppler coefficient; measurement of void coefficient in coolant; evaluation of neutron temperature; determination of neutron flux fine structure; material buckling of various lattice arrangements in water; and theoretical support studies necessary for correlation of experimental data.

Fuel Development

While the fuel design is essentially firm* and the general fabrication techniques have been described, any new fuel would require an extensive R&D effort to assure that the fabrication procedures, quality control, and

* Hanford experience with the magnesium matrix fuel has been extensive. The high corrosion rate of hot magnesium and water in case of cladding rupture make it undesirable as a fuel.
inspection procedures will provide a reasonable probability of reaching goal exposure. An in-reactor exposure program within the available time, which will not only run sample fuel elements to failure with both boiling and steam coolants (in particular the magnesium matrix element) must be conducted under operational conditions and with induced flaws so that the failure mechanism and the danger from flow stoppage may be thoroughly understood. An alternate fuels development program covering items such as a 6 to 9 percent molybdenum alloy or oxide, would be required to assure meeting startup with a workable fuel. Tentative fabrication procedures must be further evaluated. Creep, corrosion, and film forming data on the clad material would be required to assure satisfactory fuel performance and to define contamination difficulties.

Heat Transfer and Hydraulics
Full scale testing of an electrically heated fuel element mockup would be required to confirm adequacy of the design heat transfer coefficients and to determine the hydraulic characteristics of multiple fuel channels during operational, emergency, and startup-shutdown transients. Facilities for performing these studies are currently in place at Hanford, although minor modifications would be required to adapt them to the particular fuel design.

Graphite Design
The major questions in this area are being covered by the HAPO New Production Reactor. Confirmatory test and analysis would be required to verify the temperature distribution, channel rupture venting and layup questions. The accommodation of graphite contraction is less difficult in this design than in the NPR. The NPR graphite contraction research program and subsequent NPR operating experience will provide necessary design information.
Corrosion and Radioactivity Carry-Over

It is within this area that some exploratory research is required. The effect of carry-over on superheater filming, and the corrosion rate of the superheater elements must be tested to confirm the adequacy of cladding design. Deposition rates of crud and fission products in the external piping and turbine are relatively unknown for superheated steam conditions. Such information must be developed to assure proper design recognition of the problems and requirements.

Instrumentation

A program of instrument development will be required to assure functional adequacy of specific nuclear and process instruments required for the reactor system. Included in this area are rupture detection instrumentation and instrumentation to monitor coolant leakage into the graphite.

Facility Requirements for R&D Programs

Except for the fuel development program, no extensive new facilities would be required. Facilities are currently available to obtain data for studies relating to physics, heat transfer, and graphite design, although they may require minor modification. Costs of these modifications are included in the R&D costs. Facilities which may be required for development of fuel fabrication methods are not included in the above.

Mechanical and Equipment Development

In addition to the development and confirmation of specific areas of technology, several problems associated with equipment development are indicated. These may include development, testing, or evaluation of:

a. Fuel charging and discharging mechanisms,
b. Process channel assembly,
c. Critical reactor components, including control systems, and
d. Methods of containment.
## Summary of R&D Program Costs

<table>
<thead>
<tr>
<th>Category</th>
<th>Cost</th>
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</thead>
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<tr>
<td>Physics</td>
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<tr>
<td>Fuel Development</td>
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<tr>
<td>Heat Transfer &amp; Hydraulics</td>
<td>$300,000</td>
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<tr>
<td>Graphite Design</td>
<td>$300,000</td>
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<tr>
<td>Corrosion and Radioactivity Carry-Over</td>
<td>$500,000</td>
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<td>Instrumentation</td>
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<tr>
<td>Mechanism and Equipment Development</td>
<td>$1,000,000</td>
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<tr>
<td><strong>TOTAL</strong></td>
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</tr>
</tbody>
</table>
VIII. RESEARCH AND DEVELOPMENT PROGRAM - 1975 PLANT

1975 Plant Concept

The 1975 plant concept would make use of graphite as a moderator and steam as a coolant and also maintain the intent of the Russian design. Specifically, the use of an internally cooled single unit integral process tube would be incorporated as a fuel design basis.

No over-all "breakthrough" changes in the plant are considered, the main changes being evolutionary and lying in the use of improved materials, components, and process knowledge.

Without the benefit of engineering economic studies to back up the attractiveness of proposed changes, the following changes are indicated as being technically feasible:

1. Reheat cycle - The steam may be heated either directly in superheat channels or indirectly by a boiling loop exchanger. "Once-through" cooling cycles and the Loeffler cycle represent possibilities, however, both are more sensitive than the present cycle to system transients. If the control and equipment reliability requirements can be satisfied without large additional investment, the potential savings in pumping equipment and piping for "once-through" cooling could represent a significant improvement, particularly as applied to the pressure tube concepts.

2. Fuel exposure to 30,000 MWD/T - The additional control requirements and the fuel-use charges in effect at the start of design will influence the compromise. The use of plutonium-uranium oxides as well as higher temperature cladding materials, such as Inconel would be considered.
3. Steam cycle equipment simplification - A direct cycle would be utilized with carbon steel in all positions where it is compatible with temperature, excluding cladding. It is believed that it will be demonstrated that carbon steel can be shown to be compatible with the high free oxygen content in boiling water reactors. Inhibitors would be considered to assure low temperature shutdown protection. Some improvement in feedwater treatment, its costs, and turbine piping carryover decontamination contingency would be expected by operation of the 1967 plant. Plant operating and capital costs may be reduced by replacement of canned pumps with mechanical seal pumps. A cycle simplification in which boiling and superheat would be carried out in each fuel assembly would also be examined. This would require individual separation at each channel assembly and circulation augmentation by injection. If a favorable heat transfer situation may be arranged such that normal power transients may be accommodated by direct heat transfer between superheater and boiling channels, the lowered piping, emergency backup equipment and instrumentation control costs could well compensate for lower specific power and increased fuel element complexity.

4. Specific power of the reactor block would be considered up to a factor of three times the 1967 plant level. Since many costs associated with piping, heat transfer surface vs. neutron economy, fuel complexity, instrumentation, graphite temperature, steady state, transient and emergency coolant requirements, increase rapidly with specific power, the available gain in this area may be smaller than generally anticipated. An increase of specific power up to 25 to 100 percent of the present may well represent an optimum design point. Helium may be used to provide graphite temperature control, particularly if large moderator rupture vent passageways are required.
5. Containment may be replaced with a vapor suppression system utilizing reactor and pipe gallery shield as the steam vent and the building structure as a confinement vessel. It may be that the building structure would not be sufficient. In that case, a low pressure structure such as geodesic framework with non-welded plastic joints would be used to minimize confiner costs.

As a guess, the above programs could lower capital costs by perhaps 20 percent. Fuel cost improvements would probably be in line with those predicted for other solid clad fuel reactors.

Research and Development - 1975 Plant

As may be seen from the above, the essence of earlier design has been maintained, that is, an integral internally cooled fuel element capable of being used in either the superheater or boiling section with graphite moderation. The major changes are not "breakthrough" in nature, but lie in the improvement and simplification of the system and the imposition of more rigorous service conditions upon components and materials. Much of the improvement would be based on the service life and maintenance experience of the 1967 plant, particularly fuels development. The major R&D goals would be considered in the following areas:

A. **Increase in Specific Power**

1. Extend the program of heat transfer fluid flow studies, particularly in the region of transients. Both analytical and experimental work are envisioned.

2. Reduce hot spot factors through improved knowledge of the flux distribution and adjusted control patterns.
3. Study extended surface or "swirl" flow in the superheater section toward increasing the specific power of both the boiling and superheater channels.

B. Improved Fuel Life and Operating Economics

1. Increase operating temperatures by investigation of other superheater fuels and cladding alloys. Specifically, the use of oxides with Inconel cladding in the superheater section may prove to offer significant improvement.

2. Improve fabrication and inspection control to reduce failure and reduce cost. Simplify design.

3. Perform physics studies to improve nuclear control and permit higher enrichment with concurrent increase in fuel life potential. Determine the range of positive temperature coefficient associated with long exposure.

4. Develop zirconium external-internal cooled element cladding and process tubes in the boiling section to improve specific power.

C. Simplified Capital Components

1. Develop reliable sealed pumps.

2. Demonstrate direct cycle primary-superheater loop with carbon steel in ex-reactor positions.

3. Improve knowledge of radioactive carry-over to reduce size of feedwater cleanup system.

4. Distribute piping on both top and bottom faces of reactor to permit fitting simplification.
5. Design a building confinement system in which shielding structure is utilized as a channel for vapor suppression. A supported membrane outer containment system would be considered for final barrier protection if required.

D. Improved Cycle Performance

1. Increase peak temperature and turbine inlet pressure.

2. Study and develop a reheat cycle utilizing either indirect heating from the boiling loop or direct reheat in superheater elements.

To reach the aforementioned goals, R&D programs would be required as follows:

Physics

Changes in fuel element alloys, enrichment levels, or geometrical configuration will require additional test work to evaluate lattice parameters. Costs would be somewhat less than for the basic program.

Fuel Development

A continuing program of improved fuel performance would be required, using the 1967 plant as the test facility. This program would require a minimum of $500,000 per year for a four-year period.

Heat Transfer and Hydraulics

Evaluation of heat transfer and hydraulic characteristics of alternate fuel concepts would be required. Particular emphasis would be placed on heat transfer burnout and hydraulic stability as the specific power is increased by 25-40 percent. This program would require $100,000 per year over a three-year period.
Use of Carbon Steel in External Loops

The use of carbon steel in a reactor system which employs both saturated steam and superheated steam as the flowing fluid would require additional exploratory development before feasibility could be demonstrated. This program might require the development of unique but not expensive carbon steels, and evaluation of corrosion characteristics. Facilities are available for conducting these tests. Program costs would be approximately $100,000 per year over a three-year period.
REFERENCES


PLANT NET GENERATION — 300,000 KW
PLANT NET HEAT RATE —— 9,648 BTU/kW.hr
PLANT NET EFFICIENCY —— 35.4%
FUEL ELEMENT LONGITUDINAL SECTION

STN STL COOLANT RETURN TUBE 0.434" ID X 0.020" WALL
FUEL 0.946" O.D. URANIUM-MAGNESIUM MATRIX (2%MA BY WT)
STN STL CLADDING 0.008"
GRAPHITE

STN STL COOLANT SUPPLY TUBE 0.257" I.D. x 0.034" WALL

GRAPHITE BLOCK 7.97" SQ.

SECTION A-A

CUT-OUT WHEN REQUIRED FOR CONTROL ROD INSTALLATION