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SCIENTIFIC INTELLIGENCE RESEARCH AID

A TECHNICAL EVALUATION OF
THE SOVIET NUCLEAR POWER PROGRAM

CIA/SL 78-58
15 May 1958

Prepared Under Contract for
CENTRAL INTELLIGENCE AGENCY
OFFICE OF SCIENTIFIC INTELLIGENCE

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Scientific Intelligence Research Aid

A TECHNICAL EVALUATION OF THE SOVIET NUCLEAR POWER PROGRAM

NOTICE

This publication has been prepared as an aid to intelligence analysts and others concerned with the subject matter presented. Editorial review has been minimized in order to accelerate dissemination of the information.

CIA/SI 75–58
15 May 1958

For

CENTRAL INTELLIGENCE AGENCY
OFFICE OF SCIENTIFIC INTELLIGENCE

Approved for Release: 2017/05/08 C00659829
The objectives of the report are to estimate the technical feasibility of the Soviet nuclear power program as outlined in the Sixth Five-Year Plan (1956-60) and the probable course of this program in the years 1961-70. A secondary objective of the project is to estimate the effect of the Soviet nuclear power program on their stockpile of fissionable materials. The estimates presented herein are not contrary to the immediate views of the Office of Scientific Intelligence. To arrive at these estimates, each Soviet power reactor built or planned has been analyzed in detail. Soviet accomplishments in reactor physics and development of reactor materials (fuels, coolants, moderators, etc.) have been evaluated, and the cost (by U.S. standards) of the electric power produced by each type of nuclear power station is estimated. Each Soviet power reactor was also evaluated with respect to its suitability for propulsion applications.
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I. Introduction

The objectives of this report are to evaluate the Soviet nuclear power program as outlined in the Sixth Five-Year Plan (1956-1960), and to estimate the course of the Soviet nuclear power program of 1961-1970. Two useful sources of information are available as raw material for the analysis. One of these is the small collection of official or semi-official statements by officials or workers in the Soviet program. The other is the fairly extensive body of openly published technical literature. Most of this literature is now out-of-date by a year or two. The information is not of significant value in a technical study of this kind because it is too old. No useful information is available which was not knowingly released by the Soviets. It must be obvious that conclusions drawn from such information will be subject to any errors which the Soviets may have implanted either by misleading statements or simply by the withholding of information.

Despite the limitations described above, it is possible and worthwhile to examine the available information in the light of USA experience in the nuclear energy field, to determine whether there are any obvious inconsistencies in the information, and to try to interpret it as an indication of the true status of nuclear power development in the USSR. This has been done in the following report. Section II summarizes the study and states the significant conclusions.

In Section III, the program, as officially stated by the Soviets, is presented and is discussed briefly. In Section IV, forecasts are made of the probable success of the announced program through 1960 and the probable course of the 1960-1970 program. These forecasts are based on the information in the following sections, both directly and indirectly. Section V is a description, discussion, and, insofar as possible, an evaluation of the reactor types which have been stated to be under development in the Soviet Union. Reports from Soviet sources exist on the two types which are apparently receiving the most intensive development. Discussions on the other types rely heavily on USA experience with similar types. Section VI, on chemical reprocessing and waste disposal, discusses only possible developments since there is no published literature on actual Soviet work. Section VII treats the probable economics of the important reactor types from the standpoint of USA experience. Section VIII discusses the status of Soviet technology in a number of fields important to the nuclear power industry. This section is based almost entirely on technical reports in the open Soviet literature. Possible significant advances in the nuclear power field are discussed in Section IX. Appendices I, II, and III are concerned with calculations on the two most prominent types of reactors in the Soviet power program. Assumptions made in these calculations are based largely on U.S. experience in the nuclear power field.
II. Summary and Conclusions

A. General Comments on Available Data

The analysis which has been made of the Soviet nuclear power effort is based essentially on information which has been knowingly released by the USSR. The limitations of such an analysis are obvious. It can be said that no gross inconsistencies have been found between the official announcements and the published technical reports from the Soviet Union. That is not to say that the official statements should be taken at face value, but simply that they are not obviously untrue. The analysis which has been made could be seriously in error because of the withholding of information in certain areas by the Soviets. Such a withholding of information might well be expected, for we know that they attempt to time their announcements in such a way as to make them most valuable for propaganda purposes. It seems quite likely that the announcement of current accomplishments may be held until the second Geneva Conference, which is scheduled for the summer of 1956.

Within the limitations imposed by the availability of information, the conclusions summarized below have been drawn from the analysis.

B. General Status of Nuclear Technology

The available Soviet technical literature in the reactor field generally falls into one of two areas. It is either rather fundamental in nature—having to do with either theoretical subjects or with laboratory-type experiments—or it describes the actual construction or operation of a reactor which has been built. There is not much published on what one would call reactor engineering. There are, for example, a number of papers on radiation effects in uranium, but there are no papers on the practical metallurgy of uranium or of the fabrication of fuel elements. There is little doubt that these characteristics of the literature result from classification policies in the USSR, and, indeed, the United States classification policy has to some extent followed the same pattern. Perhaps because of such restrictions on publication, one gets the impression that the Soviets are well advanced in their basic scientific and technical knowledge, but that their technology may lack depth in the practical areas and in the areas of supporting technologies. Such a deficiency, relative to the United States, might well be expected, inasmuch as the United States has had several years prior start in the nuclear power field. It has been demonstrated, notably by the British, that a nation can make a late start in the nuclear power field and still make very rapid advances if it restricts its development to one or two reactor types. It is not so obvious that a country can quickly develop a broad program involving a number of reactor types with equal success. If the Soviets can meet their stated schedule with all or most of the reactor types which they claim to have under development, there will be little question that their practical technology is at least on a par with that of the United States.
In any case, it is quite obvious that the Soviet program has the benefit of extremely able scientists as well as clever and original reactor designers. Their technical literature indicates that the scientific knowledge relating to reactor development is broad and accurate, but no important information has been published in Soviet papers which was not already known in the United States.

C. Reactor Types

Of the reactor types which the USSR claims to be developing (see tables on page 8 and page 47), the water-cooled types are obviously in the most advanced stages of development. The graphite-moderated water-cooled type has been in operation as a power reactor since 1954, and a good deal of the technology of this reactor is applicable to the pressurized water type. The Russians have, however, apparently undertaken to develop zirconium for the PWR reactor. This development may be the determining factor in the actual date of operation of the pressurized water reactors. The gas-cooled, D2O-moderated reactor, although once of interest, is not in the currently-announced program. It seems most probable that development difficulties—possibly the development of suitable fuel elements—has caused its postponement. Of the reactor types scheduled as experimental stations, the boiling reactor represents the smallest departure from the main body of Soviet experience. Most of its basic problems are similar to those for pressurized water-cooled types. However, there are a number of specific design problems, and the use of natural circulation for a reactor of such high output may pose some additional problems. There is no indication of large-scale Soviet experience with sodium, applicable to the proposed fast reactor and the proposed sodium-graphite reactor. The basic knowledge of sodium, as might be obtained from laboratory-type experiments, is, however, obviously extensive. Indications are that the technology of aqueous homogeneous reactors is not highly developed. A deficiency in this respect, however, would not necessarily prevent the early construction of a small reactor of this type.

It would be reasonable to assume that Soviet technology is behind that of the United States in all of the reactor types described above except the graphite-moderated, water-cooled type, which has not been developed in the United States as a power reactor type. If the proposed design of the pressurized water reactor is successfully built, the Soviet technology on this reactor type will have overtaken that of the United States, and possibly have surpassed it.

D. Limiting Factors on Rate of Nuclear Power Development

There are three factors limiting the rate at which the nuclear power capability can grow in any country. These are the limitations imposed by technical development, limitations in the supply of special materials, and the simple capital and manpower limitations on rate of plant construction. In addition, the rate of construction of nuclear plants may be limited through choice, if the power produced is more expensive than that produced by fossil fuel plants. At the present time, all nuclear plants...
are more expensive than fossil fuel plants in the United States and, according to Soviet statements, also in the USSR. Therefore, the rate at which early nuclear plants are built in these two countries will be determined as a matter of choice by those directing the programs, and the choice will be based primarily on considerations of the contributions of the new plants to the development of the technology, and to some extent on questions of prestige. Considerations of technical feasibility will probably not enter into the question of the rate of building plants of the pressurized water or water-cooled graphite types in the Soviet Union.

Apparently only the United States is entirely free of restrictions which originate in limited supplies of special nuclear materials. The USSR apparently does not have a comfortably large supply of enriched uranium. This conclusion is inferred from the frequent mention of plutonium recycle by the Soviets, and from the fact that they have made only small quantities of enriched uranium available to other countries. Their production of D₂O is also reported to be small by USA standards (100 tons per year).

Any country which has only a small supply of separated uranium must face the question of how to fuel its proposed nuclear power plants. Britain and France have chosen to build their nuclear power industry around natural uranium, at least in its early stages. The USSR has chosen to adopt reactor types which require enriched fuel. Whether the enriched uranium supply will constitute any real restriction on the rate at which such reactors can be built and initially fueled in the Soviet Union is not clear, but it does appear that the Soviets are counting heavily on plutonium recycle for subsequent fueling of these reactors. Full reliance on plutonium recycle involves a degree of risk, and it would be expected that some other possibility would be under development as a back-up effort. The D₂O-moderated, gas-cooled reactor originally constituted such a back-up for the USSR. It would be surprising if this reactor is permanently absent from their program unless it is replaced by some other natural uranium type. If a choice must be made between refueling with enriched uranium for an indefinitely long time, or using D₂O for natural uranium reactors, then the D₂O route is probably the cheaper.*

---

* If USA prices are taken as indicative of true relative prices, the following argument can be made. D₂O costs $28 per pound and is supplied only once during the life of the reactor. 1.3% enriched uranium costs $53 per pound, as compared to natural uranium at $18 per pound, a differential of $35 per pound. This differential, or a large fraction of it, must be paid every time the reactor is refueled. Since comparable weights of D₂O and fuel are used in the reactor, the fuel differential is a much more important cost than the D₂O cost.
E. Chances For Success of Current Program

There is no direct evidence that the currently announced Soviet reactor program cannot or will not be met. It can be said, however, that if it is met, a concerted effort will have to be made by the Soviets, even if no unexpectedly large technical difficulties materialize. Since it would appear that the development of nuclear power, although certainly considered important by the Soviets, is probably not a vital necessity at the present time, it will be a little surprising if all of the proposed reactors are constructed in the 1955-1960 period. Very probably the graphite-moderated, water-cooled station will be completed in this period, and at least one of the pressurized water stations. There is probably no technical reason why both of these pressurized water stations could not be completed by 1960, but one would expect that Soviet authorities would desire sufficient lag between the first and second plants of a given type to allow the experience gained in building the first to be utilized in the second. This would mean probably a difference of at least a year in the schedule of the two plants, and perhaps two years.

In the case of experimental plants, it is generally the practice to utilize the experience obtained with the first plant for improving the second, rather than to meet a rigid schedule. It would therefore not be surprising if some of the Soviet experimental plants failed to meet their schedules. For reasons discussed in Section C above, the boiling water reactor may be the first of the experimental plants. The metal-cooled reactors (fast and graphite-moderated) will probably lag somewhat more. Of the two, more emphasis may well be put on the fast reactor. Because of its small size, the homogeneous reactor could conceivably appear at any time, depending on a decision as to how far to carry laboratory development before building a small reactor.

F. Probable Course of Program to 1970

The apparent Soviet system of building quite large demonstration reactors of a given type as soon as the technology warrants, is deemed to be an effective one, and one well adapted to their governmental setup. It will probably be continued. This would mean that the number of reactors built may be large for a given total generating capacity. Assuming Soviet "first round" power stations will come within 50% of meeting the Russian price of power produced by fossil fuels, it could be that the "second round" pressurized water reactors and/or water-cooled graphite reactors might approach closely the goal of competitive nuclear power in the USSR, by virtue of larger capacity and experience gained with the "first round" reactors. It might be expected that the "second round" reactors would go into operation by 1965, that there would be between 5 and 8 stations, and that the output per station would be perhaps in the 600 eMW (electrical megawatt) range. If such stations are truly competitive, the rate of construction could conceivably increase sharply after 1965, provided there is no limitation imposed by the supply of fissionable material. It is characteristic of the fossil fuel power industry that although it can be expanded rapidly in a given area where supplies of coal exist, or where transportation has already been established, its expansion into remote
or newly-settled areas can involve large capital investments. In such areas it may be more economical to establish a nuclear power industry.

In the years immediately following a decision to adopt nuclear power as a major power source, the limitation to rate of expansion may lie in such things as limited technical manpower and limitations of support facilities such as fuel fabrication and reprocessing plants. It seems improbable for these reasons that the construction of new nuclear power capacity in the 1965-1970 period could be much greater than about twice that in the preceding five years. During the entire period, from the present to 1970, large plants of the types which are now in the experimental stage may be constructed as the performance of the experimental plants indicates sufficient development. It seems improbable that any of the currently mentioned experimental types other than possibly the boiling reactor will constitute a large fraction of the total generating capacity up to 1970, unless the current large station types prove definitely uneconomic and the total capacity constructed falls much below that estimated above.

G. Areas in Which the USSR May Make Significant New Developments

Two possible areas in which Soviet developments may occur which are of relatively major importance are the development of a high-temperature water and steam-cooled reactor and the development of a reactor for export which may be significantly cheaper (relative to other commodities) than those produced by other countries. These possibilities, and others, are discussed at some length in Section IX.

H. Propulsion

The propulsion reactor for the Soviet icebreaker, the "Lenin," may be either the pressurized water reactor or the water-cooled graphite reactor. In all probability the Soviets possess the technology for producing effective submarine and ship propulsion reactors, and very probably nuclear submarines are being developed or built. There is no real indication of an advanced project in the aircraft or nuclear rocket field, The nuclear rocket seems a more likely probability for Soviet development than the airplane.

A nuclear locomotive is technically feasible, but is of little interest in the United States. It may have applications in the USSR, and if so, it may be expected as a Soviet development.

III. Soviet Statements of Nuclear Power Program, 1956-60

The original power development program proposed by the Soviet Union for their sixth Five-Year Program (1956-1960), as announced in 1956, 0-5-6, 5-1-1 called for a total of 2000-2500 MW of electrical capacity in five power stations. Apparent breakdown of the planned power division is
shown in Table I.* In order to meet the announced 2000-2500 eMW capacity, each station of Table I must have been planned near the maximum of the published range. The present plans, as published by the USSR S-1-8, S-1-9 (see Table I), apparently represent a retrenchment in three respects. First, each major power station is now scheduled to be built, at least initially, with the minimum of the previously planned power level. Secondly, one power station, the D2O-moderated, gas-cooled reactor, has apparently been postponed, at least temporarily. Third, as a result of the other two decisions, the total planned nuclear power has been reduced to about 1500 eMW. According to the reply of the USSR to the UN questionnaire, S-1-8 there will be a total of about 28,300 MW electrical generating capacity in 1960. The nuclear program will therefore constitute about 3.5% of the total Russian generating capacity. (This compares with an estimated 160,000 eMW capacity for the USA with about 0.5% being nuclear power.)

According to Kurchatov, S-1-1 "the atomic power stations will be put into operation beginning with the end of 1958; some of them will start operation in 1959 and in 1960" (see Table II). In May of 1956, U-8 the Soviets announced the beginning of construction of a 400 eMW power station, presumably the graphite-moderated, water-cooled reactor (Station 2, Table I), scheduled to be completed by mid-1959 or earlier.

**TABLE I - SOVIET POWER PROGRAM (1956-1960)**

<table>
<thead>
<tr>
<th>Station</th>
<th>Proposed 1956a (eMW)</th>
<th>Location</th>
<th>Apparent Present Plans (eMW)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. H$<em>{2}$O-moderated, H$</em>{2}$O-cooled</td>
<td>420-630</td>
<td>Urals</td>
<td>420</td>
</tr>
<tr>
<td>2. Graphite-moderated, H$_{2}$O-cooled</td>
<td>400</td>
<td>Moscow</td>
<td>400</td>
</tr>
<tr>
<td>3. Same as No. 1</td>
<td>420-630</td>
<td>Urals</td>
<td>420</td>
</tr>
<tr>
<td>4. D$_{2}$O-moderated, Gas-cooled</td>
<td>100-400</td>
<td>?</td>
<td>Postponed</td>
</tr>
<tr>
<td>5. Experimental Station of 4 Prototype Reactors</td>
<td>175</td>
<td>?</td>
<td>175</td>
</tr>
<tr>
<td>6. Ship Propulsion</td>
<td>60</td>
<td>Leningrad</td>
<td>60</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>1575-2293</strong></td>
<td></td>
<td><strong>1475</strong></td>
</tr>
</tbody>
</table>

a It is conceivable that an additional power reactor of the PWR type, utilizing enriched U-235 and thorium, may have been considered for construction, and later postponed.

* In the remainder of this report the individual power stations will be referred to by the number in the first column of Table I.
<table>
<thead>
<tr>
<th>Reactor and Type</th>
<th>Heat (MW)</th>
<th>Electrical (MW)</th>
<th>Efficiency</th>
<th>Specific Power</th>
<th>Steam Conditions</th>
<th>Type</th>
<th>Amount Used</th>
<th>Annual Burnup</th>
<th>Conversion Ratio</th>
<th>Amount Proc. Actually</th>
<th>Fuel Element Characteristics</th>
<th>Estimated Total</th>
<th>Location</th>
<th>Estimated Date of Operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>1) Power Station, Project 1 water mol., water cooled</td>
<td>1500</td>
<td>1500</td>
<td>25.5%</td>
<td>357.6</td>
<td>RH at 1050°F</td>
<td>1.1%</td>
<td>40%</td>
<td>600 kg</td>
<td>0.65 tons</td>
<td>111 kg</td>
<td>None in Er coating Poor fuel probably in cladding rather than SS</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2) Power Station, Project 2 graphite mol., water cooled 2 reactors per station</td>
<td>1500</td>
<td>1500</td>
<td>25.5%</td>
<td>357.6</td>
<td>RH at 1050°F</td>
<td>1.1%</td>
<td>40%</td>
<td>600 kg</td>
<td>0.65 tons</td>
<td>111 kg</td>
<td>None in Er coating Poor fuel probably in cladding rather than SS</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>3) Power Station, reactor water cooled</td>
<td>100-200</td>
<td>100-200</td>
<td>25.5%</td>
<td>357.6</td>
<td>RH at 1050°F</td>
<td>1.1%</td>
<td>40%</td>
<td>600 kg</td>
<td>0.65 tons</td>
<td>111 kg</td>
<td>None in Er coating Poor fuel probably in cladding rather than SS</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>4) Experimental Beryllium</td>
<td>100</td>
<td>100</td>
<td>25.5%</td>
<td>357.6</td>
<td>RH at 1050°F</td>
<td>1.1%</td>
<td>40%</td>
<td>600 kg</td>
<td>0.65 tons</td>
<td>111 kg</td>
<td>None in Er coating Poor fuel probably in cladding rather than SS</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>5) Experimental Graphite mol., Na cooled</td>
<td>150-100</td>
<td>100</td>
<td>25.5%</td>
<td>357.6</td>
<td>RH at 1050°F</td>
<td>1.1%</td>
<td>40%</td>
<td>600 kg</td>
<td>0.65 tons</td>
<td>111 kg</td>
<td>None in Er coating Poor fuel probably in cladding rather than SS</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>6) Experimental Graphite mod., Na cooled</td>
<td>200</td>
<td>200</td>
<td>25.5%</td>
<td>357.6</td>
<td>RH at 1050°F</td>
<td>1.1%</td>
<td>40%</td>
<td>600 kg</td>
<td>0.65 tons</td>
<td>111 kg</td>
<td>None in Er coating Poor fuel probably in cladding rather than SS</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>7) Experimental Neutron</td>
<td>25-15</td>
<td>15</td>
<td>25.5%</td>
<td>357.6</td>
<td>RH at 1050°F</td>
<td>1.1%</td>
<td>40%</td>
<td>600 kg</td>
<td>0.65 tons</td>
<td>111 kg</td>
<td>None in Er coating Poor fuel probably in cladding rather than SS</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>8) Experimental Neutron</td>
<td>25-15</td>
<td>15</td>
<td>25.5%</td>
<td>357.6</td>
<td>RH at 1050°F</td>
<td>1.1%</td>
<td>40%</td>
<td>600 kg</td>
<td>0.65 tons</td>
<td>111 kg</td>
<td>None in Er coating Poor fuel probably in cladding rather than SS</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>9) Experimental Neutron</td>
<td>25-15</td>
<td>15</td>
<td>25.5%</td>
<td>357.6</td>
<td>RH at 1050°F</td>
<td>1.1%</td>
<td>40%</td>
<td>600 kg</td>
<td>0.65 tons</td>
<td>111 kg</td>
<td>None in Er coating Poor fuel probably in cladding rather than SS</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Legend: MW, megawatts; kV, kilovolts; kg, kilograms; l, liter; atm, atmosphere pressure; ton, tons; (T) refer to extra tons (1000 lb.)
<table>
<thead>
<tr>
<th>TIME (Minutes)</th>
<th>00</th>
<th>10</th>
<th>20</th>
<th>30</th>
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**Table 1:**

Characteristics of Power takeover from:

Date: [date]

**Notes:**

- Power takeover:
- Generator rating:
- Load requirements:
- Efficiency:
- Specific gravity:
- Horsepower rating:
- Oil type:
- Lubrication:
- Oil temperature:
- Coolant temperature:
- Ambient temperature:
- Barometric pressure:
- Altitude:
- Type of oil:
- Type of fuel:
- Type of lubricant:
- Type of coolant:
- Type of atmosphere:
- Type of insulation:
- Type of bearing:
- Type of seal:
- Type of construction:
- Type of protection:
- Type of insulation:
- Type of bearing:
- Type of seal:
- Type of construction:
- Type of protection:
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- Type of construction:
This type of power station proposed by the Soviets is a two-reactor, 400 MW (electrical) station near Moscow. The reactors in this station are to be graphite-moderated, water-cooled thermal reactors S-1-9 of the Atomic Power Station (APS) type reported at the Geneva Conference, S-3-6, and in the Soviet Journal of Atomic Energy. S-2-2. The Geneva paper reported that the Russians expected to be able to attain a conversion ratio of 0.5 in future power stations of the APS type. Hence, this value of the conversion ratio has been assumed for the calculations reported herein, although it is not the ultimate value that might be attained with this reactor type (see Section IV-3). The new power station, though patterned after the APS-1, S-3-6 may use zirconium-jacketed metallic uranium fuel rather than the stainless steel-jacketed fuel of APS-1, with a resulting improvement in enrichment and/or conversion ratio. It is also stated that the steam conditions will be much better than for APS-1.

The two IFR-type power stations, originally scheduled for construction in the Urals, were apparently to have been of a total of about 1,000 MW electrical capacity, S-5-6, and of the pressurized, water-moderated, water-cooled type. On the basis of the reported S-1-9 characteristics of the Soviet FWR, this would have corresponded to three reactors of 210 MW (electrical) capacity in one power station and two reactors in the second station. However, it is quite possible that the described IFR reactor may prove to be capable of a power output higher than its present rating. At the Belgrade conference and again in the reply to the UN questionnaire, the power stations were stated to be of 420 MW each, with two reactors in each station (see Table 1), thus apparently representing a decrease in the planned net power output of the two stations.

In June 1957, S-5-7 the newspaper "Tass" announced that the Soviets had begun construction of a 420 eMW power station, with two reactors, each driving three 70 MW turbines. This is apparently the first of the two IFR-type, water-moderated, water-cooled reactor stations scheduled to be completed in 1960, probably in the early part of the year.

The proposed ultimate method of operation of the Soviet FWR is to recycle the produced plutonium, mixed with natural uranium for subsequent fuel loadings. With the conversion ratio expected, the plutonium produced would be sufficient to increase the effective enrichment of natural uranium to the point where it would be satisfactory for use as the reactor fuel. Thus the reactor is "self-sustaining" with respect to the breeding of sufficient plutonium to combine with the U-235 already present in natural uranium, rather than with respect to the breeding of as much plutonium as U-235 burned. It should be noted that the possibility of using recycled plutonium for "enrichment" of natural uranium is one which is hoped for by every country which desires a large nuclear power capacity, but which lacks large facilities for separation of the uranium isotopes. The main obstacle to such use of recycled plutonium is not the achievement of a conversion ratio and enrichment value which are compatible with a "self-sustaining" system, but the technical problems of developing and fabricating fuel elements which contain recycled plutonium. There is no evidence that Russia has solved the problem of achieving such operation economically, or otherwise.
Kurchatov's interest in this type of reactor, and his apparent preeminence in the Soviet hierarchy, indicates that in the near future the USSR reactor program may rely heavily upon the pressurized water reactor. S-1-1 Small size, high specific power reactors of the PWR type apparently appeal to the Soviets in that small installations are required. This type of reactor is also quite suitable for propulsion, at least of marine vessels. The PWR would not be suitable for aircraft propulsion in any form recognizably like its present one.

A major modification of the PWR reactor, which is apparently highly favored by Kurchatov as a long-range possibility, is the highly-enriched uranium, thorium-fueled PWR-type reactor. In his talks, Kurchatov has said that it may be possible to operate this type of reactor with a positive breeding gain, i.e., to produce as much fissionable material as is burned, and the reactor would therefore be self-sustaining with no additional fuel required. S-1-1 Such a thorium reactor would be similar to the Consolidated Edison Reactor being built in this country, but with greatly improved breeding characteristics. A further incentive to the development of this type reactor, might be or at least to the exploitation of its propaganda value, the recent announcement U-10 that a "large deposit of uranium and thorium ore lying within 10 feet of surface soil has been discovered in the Ranchi plateau" in India.

A third type of power station originally proposed for the Sixth Five-Year Plan is a D2O-moderated, gas-cooled reactor. S-1-4, S-2-1 In 1956, this reactor was slated for construction. S-5-6 The complete absence of any mention of this type of reactor in later announcements, S-1-8, S-1-9 coupled with Alikhanov's S-1-4 obvious effort to minimize the quantity of "high cost" heavy water in his design, would indicate that the high cost (or low availability) of heavy water in the USSR requires at least the postponement of this type in their current power reactor program. Another possibility, perhaps a more probable one, is that development of the reactor type has proven more difficult than anticipated. The development of a high temperature, low absorption jacket for fuel elements would be a particularly difficult objective to achieve during the initially indicated time interval. Because of the general attractiveness and desirability of this type of reactor, it will probably be replaced on the power development program when the Soviet value for heavy water is reduced sufficiently to allow its use in power stations. Construction of this type of power station may begin in the 1956-1960 period, with completion expected sometime in the middle of the seventh Five-Year Plan. A recent announcement in "Chemische Industrie" U-11 reports that Russia now has two plants producing heavy water. One plant has a capacity of 60 tons per year and the other of 30 to 40 tons per year, making a total production of 90 to 100 tons of heavy water per year. This amount of production would not as yet be large enough to lower the cost of heavy water sufficiently to permit economic use in large-scale reactors, and the USSR, for political reasons, would probably not consider purchasing the required supply. However, the large reserves of cheap hydroelectric power (1,700,000 kWh per year S-1-8) indicates that the potential power reserve is sufficient to provide the necessary D2O production in years to come, and to reduce the cost of D2O.
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Some question exists as to whether the fifth power station originally scheduled in the Sixth Five-Year Plan was to be of an unspecified type, or whether the four prototype reactors, lumped together, were considered as a single power "station." Ye. P. Slavsky, S-1-5 in an interview reported in "Izvestiya," reported that "an experimental atomic power station with a capacity of 200,000 kw will be built. This station will be equipped with four other type reactors." Whether all four experimental prototype reactors will be built in the same location is not known. A recent announcement S-5-4 reported the beginning of construction of a research reactor in Georgia, which could conceivably be one of the experimental types. Table II summarizes the available information on the four types of experimental reactors.

A considerable effort will probably be expended on the construction of at least the three major power stations (the two pressurized-water stations and the water-graphite station). With a maximum effort, a period of at least two years per station would be required to construct power stations of the magnitude described. If construction of the remaining power station (Station No. 3, Table I) is begun in 1958 as scheduled, all three should be operating by the end of 1960 or the early part of 1961. Thus, the current program, as established by the Soviets, appears to be quite reasonable, and there is no direct evidence that they will not be able to meet the planned schedule.

IV. Projected Program

Although the nuclear reactor program in the USSR appears to follow closely the general pattern of that in the USA, it would be unwise to think that the Soviets are simply copying our program and developments, rather than proceeding competitively on their own parallel development. However, for prestige reasons, our program probably will influence greatly the magnitude of their future reactor program, as well as requiring that they investigate those experimental reactors which show promise in this country. According to Bertrand de Jouvenel, a French economist, U-36 "The avowed purpose of the Soviet government is to bring Russian industrial power, in the shortest possible time, to parity with that of the United States." It is obvious from the published program for the sixth five-year period that, with the exception of the experimental reactors, their power stations are all planned to be larger in size than the analogous US plant. Apart from the propaganda value of such a program, the power costs should be lower for plants of increased power level. In addition, there is the apparent strong motivation in the USSR to produce as much plutonium as possible for military purposes.

Future developments in the succeeding two five-year programs (1960-1970) will probably follow the same general trend, marked by larger scale building of a fewer number of reactor types than in the USA. It would appear that the Soviet program has been modified and that future developments will be inclined toward heavy utilization of water-moderated, water-cooled PWR type reactors. Therefore, the seventh and eighth five-year programs will probably see the construction of a large number of PWR type reactors, both for large power plants as well as for ship propulsion.
Of the smaller scale experimental reactors, little can be said of their expected use by the Soviets in their seventh and eighth five-year programs, since the results of their operation, and that of similar types in the USA, will influence the degree of emphasis placed on their use in large power plants. Soviet interest in fast reactors and their potentially high conversion ratio may mean that this type of reactor (now experimental) may appear in coming five-year programs as an important large-scale power station. The next ten years may see the construction of several additional types of small-scale experimental plants, such as Be or BeO moderated, forced circulation, homogeneous, and fused salt homogeneous reactors.

Although the USSR has vast sources of raw material, S-1-8 the majority (70%) of their fuel reserves are in Siberia. The cost of electric power in the more populated western areas is considerably influenced by the high cost of long-haul fuel. In the Moscow area, electricity costs between 6 and 10 kopeks/kwh as compared to 4 - 7 kopeks/kwh in the Siberian areas. S-1-8 Again assuming Soviet nuclear power to be 1-1/2 times the equivalent cost of coal-produced electricity, its cost would be about 15 kopeks/kwh. On the basis of the published rate of exchange (1 kopek equals 2.5 mills), this would correspond to a power cost of 37.5 mills/kwh. This is about twice the anticipated power cost in advanced optimized nuclear plants (not presently constructed plants) in this country (19.4 mills/kwh for a homogenous plant; U-2 19.6 mills/kwh for the PWR; U-12 and 17.9 mills/kwh for a Calder Hall type U-12).

Soviet power costs are computed on the basis of a plant factor of 0.85 (stated that "stations are guaranteed a base load utilizing installed capacity for a period of 7500 hours per year" S-1-9). This represents a fairly large plant factor for other than the beginning of the plant lifetime, but may be justified in regions where power is badly needed and may be properly scheduled. The remaining 15% of the time should be sufficient for maintenance of the plant and necessary repairs which are of such a nature as to require shutdown.

Although the Soviets state that they believe their power stations are inherently safe and reliable, they are still being located a considerable distance from cities and large populated centers. Future use of the process steam for industrial and residential heating has been apparently studied, but rejected for the present time until additional experience with large plants has been gained.

If the published expected total power generation in the USSR S-1-8 for the years 1957 (211 x 10^9 kwh or 28,300 MW capacity) and 1960 (320 x 10^9 kwh or 42,000 MW capacity) are projected to the year 1970 (Figure 1), the anticipated total power capacity in Russia would be about 140,000 MW. The percentage rate of this projected growth is somewhat greater than that anticipated in the USA or UK. However, their intensified government-controlled drive to increase the industrial capacity in the USSR would justify the greater relative increase, at least through the year 1970. At the end of 1960 or 1961, the Russians should have completed their modified Sixth Five-Year Plan, and the installed nuclear power at that time will
Figure 1
Projected Power Development
1957 - 1970
MW Installed Electrical Capacity

British Nuclear Program (U-9)

Project USSR Nuclear Program

USA Nuclear Power (U-7)

Estimated Total Power - USSR

10,000
1,000

10^5

Installed Nuclear Power Capacity

Total Electrical Capacity (MW)

6th 5-year Program
1957 59 61 63 65 67 69
100

7th 5-year Program
1960 62 64 66 68 70

8th 5-year Program
1965 67 69 70

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comprise about 3 - 3.5% of the total. (USA value is 0.5%) Assuming that by 1970 the nuclear power capacity will quite likely have reached 5 - 10% of the total power generating capacity (estimated USA value is 6.3%; UK is about 33%), then an installed nuclear capacity of 7,000 to 14,000 MW may be expected. Figure 1 shows the estimated nuclear power growth in the USSR for the years 1959 to 1970. For comparison, the projected UK program, as announced in their reply to the UN, U-9 and the USA program, as estimated by Davis and Roddis, U-7 are also shown. The estimated USA development in nuclear power is, however, not based upon a nuclear program, but is an estimate of what the USA growth might be in coming years.

Based on this projected power growth, the number of power stations, if built as 400 MW plants, of two reactors each, would be 17 - 35 (35 - 70 reactors) by 1970. If a specific power of about 30 MW/T average is assumed, there would then be 700 to 1400 tons of fuel in use. However, future development, as a consequence of the reduced specific cost of the higher power reactors, will be directed toward the construction of larger and larger power stations so as to claim the attendant economic advantages. The Soviet nuclear power program in future years (1960-1970) will most likely be a compromise between the USA power development program and that in England. The USSR program will probably concentrate heavily on a large-size single type reactor station (most likely the PWR type), resembling the UK program, but with several stations of other types (as in the USA), such as gas-cooled, heavy water-moderated, water-cooled, graphite-moderated, or other types of experimental plants which may show economic possibilities for the future.

In order to estimate the use of nuclear fuels in the years 1960-1970, it is necessary to make a number of assumptions concerning the average reactor characteristics and to estimate the effects of future technological developments. First, it is assumed that fuel burnup will be advanced from its present 3500 MW/T to 10,000 MW/T by 1965, and will reach 15,000 MW/T by 1970. The gradual increase assumed for this quantity is shown in Column 3 of Table III. These are average burnup figures which may quite reasonably be attained within this period of time. Secondly, the published plant factor of 0.85 claimed by the Soviets is assumed to be correct and to be effective for all the nuclear plants operating in 1960-1970. Third, an average enrichment of 1.5% has been assumed for these calculations. This figure is somewhat higher than the quoted figure for the Soviet PWR and was so selected to compensate for increased reactivity required for longer burnups, and for the possible use of thorium in some of the reactors.

Fourth, it was assumed that the plants will be able to operate at an efficiency of 29% and with an effective conversion ratio of 0.85. These figures represent reasonable average values, commonly considered to be readily attainable in a few years. The present Soviet plans for their PWR reactor anticipate an efficiency of 25.5% and a conversion ratio of 0.8. Fifth, an average specific power of 10 eW/T was assumed in order to compute the tonnage of uranium required for the USSR's power development program. This is about twice that for the Soviet PWR reactor, and reflects the possible improvements in future years.
In Table III, and also in Figures 2 and 3, are shown the results of the estimated Soviet power development program on their use and requirements of fissionable material. For these calculations, the average of the projected nuclear power program range was used. In Columns 5 and 6 of Table III, and in Figure 2, are shown the calculated annual burnup of U-235 and production of plutonium. For these calculations, a delay of one year was allowed for the time that the fuel is in the reactor or processing plant, before the plutonium is actually available. It is interesting to see what the effect would be if the Soviet power program were to be used principally for plutonium production, at a sacrifice of economic power production. The maximum plutonium production potentially available from the projected reactors is shown in Column 7 of Table III and in Figure 2. A delay of 6 months was allowed for operation and handling in computing this curve. Such a method of operation would place greater emphasis on plutonium than on power production, and would require shorter fuel life-time (so as not to burn the plutonium produced), higher consumption of U-235, and larger reprocessing facilities. In view of the Soviet drive toward industrial growth, such a method of operation would not be likely, except in the event of a war which would place a large premium upon plutonium for military use.

Shown in Column 8 of Table III and in Figure 3 is the calculated annual requirement of replacement fuel (1.5% initial enrichment). Since these figures represent the tonnage of fuel being replaced annually, they also represent the net capacity of chemical reprocessing plants required to handle the projected Soviet power program.

Adding the annual requirement for new construction to that required for replacement, the total annual fuel requirement is shown in Column 10 and in Figure 3. Since about 2.9 tons of natural uranium are required for the production of 1 ton of 1.5% enriched uranium (assuming that the tailings are rejected at 0.3% U-235), then the annual requirement for natural uranium (Column 11, and Figure 3) is about 2.9 times the annual requirement for fuel. These figures also represent the capacity of diffusion (or other isotope concentrating) plants required to process the natural uranium into the necessary fuel.

If the plutonium produced (Column 5 and Figure 2) is recycled and added to natural uranium to constitute the makeup of enriched fuel, then Column 10 (tons of fuel required annually) becomes equal to the tons of natural uranium required annually, and the diffusion plant capacity would be drastically lowered. However, the plutonium produced will likely be required for military purposes, and hence would not be used for recycle for a number of years.

In these estimations, the possible use of thorium as the fertile material has not been included. Due to the difficulty of handling and processing the U-233 produced from thorium, it can only be guessed that the use of thorium could possibly reach as much as 10-50% of that of uranium, although there are no indices on which to base a reliable estimate of the expected use of thorium by the Soviets. In any event, the requirement for U-235 would not be appreciably changed in the early part of the develop-
Figure 2

Annual Consumption and Production of Fissionable Material for
Projected Soviet Nuclear Power Program

1957 - 1970

Maximum Available Plutonium Production

Fissionable Material Consumed Annually

Estimated Fissionable Material Produced Annually

10,000

1,000

Kilograms

58 59 60 61 62 63 64 65 66 67 68 69 70

5-year Program 5-year Program 5-year Program

1957 1960 1965 1970

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### TABLE III

**PROJECTED SOVIET NUCLEAR POWER PROGRAM IN THE USSR**

<table>
<thead>
<tr>
<th>Beginning of Year</th>
<th>Average of Projected Capacity (MW)</th>
<th>Fuel Burnup (MWD/T)</th>
<th>Burndown of U-235 at Conversion Ratio of 0.85</th>
<th>Maximum Available Production of Pu</th>
<th>Annual Production of Pu</th>
<th>Annual Requirement of Reproduction Fuel</th>
<th>Annual Requirement of New Construction for Fuel</th>
<th>Total Annual Requirement for Fuel</th>
<th>Total Annual Requirement of Enrichment with no Recycle</th>
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<td>1,220</td>
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<td>631</td>
<td>137</td>
<td>768</td>
</tr>
<tr>
<td>1970</td>
<td>10,500</td>
<td>15,000</td>
<td>77.4</td>
<td>8,670</td>
<td>7,360</td>
<td>12,400</td>
<td>750</td>
<td>195</td>
<td>945</td>
</tr>
</tbody>
</table>

---

*a* Assuming plant factor of 0.85; efficiency of 29%; enrichment of 1.5%.

*b* From unpublished General Nuclear Engineering Corporation curves.

*c* For high priority on plutonium production, such as war-time conditions.

*d* Also equal to annual reprocessing plant capacity required, and to annual requirements for natural uranium with plutonium recycle, as replacement fuel.

*e* Also equal to annual diffusion plant capacity required.
Figure 3
Annual Requirements of Fuel and Natural Uranium for
Projected Soviet Nuclear Power Program
1957 - 1970

Annual Requirement of Natural U
Also equal to annual diffusion plant capacity required

Total Annual Requirement for Fuel

Annual Requirement for Replacement Fuel. Also equal to Annual Reprocessing Capacity Required

Tons (Metric)

1957  58  59  60  61  62  63  64  65  66  67  68  69  70
6th  5-year Program  7th  5-year Program  8th  5-year Program

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ment, since not enough U-233 would have been produced. Hence, any thorium used would be in addition to the annual requirements of uranium as shown in Table III and Figure 3. Later use of thorium and produced U-233 toward the end of 1960-1970 could decrease the annual requirement of uranium ore to some extent, and would reduce the production of plutonium, since U-233 would be produced instead. However, this possibility is accounted for by specifying Column 6 of Table III as fissionable material produced, which would include both plutonium and U-233. Excluding the possibility of a war and the resultant effects upon the value of plutonium, and considering the thorium reserves in the USSR, the Soviets may be expected to make extensive use of thorium as a fertile material in the later years, although the degree of utilization will depend largely upon technical advances in chemical processing and fuel fabrication of U-233 and thorium.

V. Evaluation of Reactor Types

A. Pressurized-Water Reactor (PWR)

The construction and development in the USSR of atomic power stations utilizing pressurized-water type reactors represents an important portion of the effort in the USSR Sixth Five-Year Plan (1956-1960). Of the five large industrial atomic power stations planned, two are to be of the pressurized-water type. It would appear, on the basis of the Belgrade report, S-1-6, S-1-9, that the pressurized-water reactor stations are furthest along in detailed engineering design. However, it is possible that the Soviets gave more attention and detail on this reactor type, knowing that several large reactors of this type are under construction or design in the USA, so that essentially no new information was divulged by their announcements.

Whenever a country commits a large portion of its nuclear power program to H2O-moderated reactors, it must face the question of how it will supply the reactors with the slightly enriched uranium which they require for operation. Although the Soviet Union has facilities for separating the uranium isotopes, it is unlikely that they presently are adequate to support a large program of operation with slightly enriched reactors. The Soviets have stated that they expect their pressurized water reactors ultimately to provide their own enrichment in the form of recycled plutonium. To accomplish this objective, the required enrichment for the reactor must be kept relatively low and the conversion ratio must be made relatively high. There is little doubt that it is technically feasible to design such reactors if zirconium is used as the structural and fuel jacketing material. The quoted enrichment and conversion ratio of the Soviet pressurized water reactor seem to be compatible with such operation. The initial fuel loading for the reactor will consist partly of slightly enriched (1.5%) uranium and partly of natural uranium. The reactor is expected to have sufficient reactivity for a fuel lifetime of 3,500 MWd/ton. The conversion ratio is quoted as 0.80.

Estimates are made in Appendix II of the physics characteristics of the Soviet PWR based on the fuel element and lattice information given in the Belgrade report, S-1-9. The calculations indicate that with fuel en-
enrichment close to the quoted value, the reactor can run to the quoted MWd/ton uranium. A conversion ratio of 0.8 or greater should be attainable with the reactor composition described in the Belgrade report.

The possibilities, nuclearwise, of high conversion ratios and plutonium recycling have been recognized and worked on in the USA. However, the economic feasibility of this mode of operation is closely related to the costs of recovering the discharged plutonium at the end of each cycle and of refabrication of the fuel elements. In the USA, the anticipated high costs of spent fuel element reprocessing is one of the major obstacles to achievement of economic nuclear power in reactors having relatively short lifetime per cycle but high conversion ratio, as do the Soviet PWR reactors. No mention is made by any of the Soviet literature or discussions by Soviet scientists of reprocessing methods and costs.

A brief description of the Atomic Power Station, as presented in the Belgrade report, follows. (See Figure 4.) The first section of the electric station (indicating that expansion of the power capabilities of the station is planned) has a total electric power capacity of 420 MW in two reactor units (210 MW/unit). The reactor is cooled and moderated with ordinary water at a pressure of 1470 lb/sq in. The pressurized water flows from the reactor to a steam generator where it gives up its heat to water flowing in a second circuit. The water in the second circuit is changed to steam in the steam generator, and is then directed to the steam turbines. The fuel elements are of uranium dioxide jacketed within zirconium tubes (see Figure 5); the initial charge is to be made of 17 tons of natural uranium dioxide and 23 tons of 1.5% enriched uranium dioxide. The Soviets indicate that a certain amount of experimentation with various fuel charges will be undertaken to determine the most "efficient" charge. This experimentation will of necessity proceed over a period of many years, inasmuch as determining the capabilities of a single initial charge through some stage of Pu recycling would involve several years of power operation.

Reactors of the pressurized-water type have been under development in the USA for some time. An idea of the size and power-producing capability of the Soviet reactor compared with the major U.S. reactors under construction is given in the following Table IV.
TABLE IV - STATED POWER, MW

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Net Reactor Heat</th>
<th>Net Electrical</th>
<th>System Pressure</th>
<th>Outlet Water Temperature From Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Russian PWR</td>
<td>760</td>
<td>195</td>
<td>1470 psia</td>
<td>527 F</td>
</tr>
<tr>
<td>Shippingport PWR U-26</td>
<td>230</td>
<td>60</td>
<td>2000 psia</td>
<td>538 F</td>
</tr>
<tr>
<td>Consolidated Edison U-27*</td>
<td>500</td>
<td>130</td>
<td>1500 psia</td>
<td>510 F</td>
</tr>
<tr>
<td>Consolidated Edison U-27**</td>
<td>500</td>
<td>219</td>
<td>1500 psia</td>
<td>510 F</td>
</tr>
<tr>
<td>Steel-Jacketed Design U-28***</td>
<td>482</td>
<td>134</td>
<td>2000 psia</td>
<td>524 F</td>
</tr>
</tbody>
</table>

Steam Pressure At Full Load

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Steam Pressure</th>
</tr>
</thead>
<tbody>
<tr>
<td>Russian PWR</td>
<td>470 psia</td>
</tr>
<tr>
<td>Shippingport PWR</td>
<td>600 psia</td>
</tr>
<tr>
<td>Consolidated Edison</td>
<td>405 psia</td>
</tr>
<tr>
<td>Consolidated Edison</td>
<td>355 psia</td>
</tr>
<tr>
<td>Steel-Jacketed Design</td>
<td>500 psia</td>
</tr>
</tbody>
</table>

Steam Temperature At Turbine Inlet

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Steam Temperature</th>
</tr>
</thead>
<tbody>
<tr>
<td>Russian PWR</td>
<td>461 F</td>
</tr>
<tr>
<td>Shippingport PWR</td>
<td>486 F</td>
</tr>
<tr>
<td>Consolidated Edison</td>
<td>446 F</td>
</tr>
<tr>
<td>Consolidated Edison</td>
<td>1000 F</td>
</tr>
<tr>
<td>Steel-Jacketed Design</td>
<td>467 F</td>
</tr>
</tbody>
</table>

* Without oil-fired superheater

** With oil-fired superheater

*** Reputed to be the conditions for the Yankee Power Plant Design.

The table shows that the reactor heat output of the Soviet PWR is about 1.5 times as large as the largest of the USA reactors under construction. A check of the heat-transfer characteristics of the PWR, on the basis of the information given in the Belgrade report, indicates that the average heat flux in the reactor (Btu/hr-sq. ft.) is comparable to USA values attained in reactor designs. The larger heat output is the result of the larger core size used in the Soviet PWR.

At the present time in the USA, the limitation on core size has been considered to be the size of the reactor pressure vessel required to hold the reactor core. In the USA, present reactor vessels to hold pressures of about 1500 lb/in² are not larger than about 9 feet in diameter, whereas the Soviet PWR is to have a 12.5 foot diameter pressure vessel. In the Belgrade report, it is stated that the pressure vessel is to be constructed
of a high-strength heat-resistant steel having a yield point of 71,000 psi at a temperature of 617°F for which the design stress used is half the yield point value, or 35,500 psi; up to the present in the USA, low carbon steel is being used in pressure vessel fabrication for which the design stress is 20,000 - 27,000 psi. According to the Soviets, the use of the high-strength steel keeps the weight of the pressure vessel down to 170 tons (without top lid) which was an important factor in being able to transport the vessel from factory to reactor site. Transportability of pressure vessels may be one of the important limitations on pressure vessel size; the actual physical dimensions of the vessel may ultimately be a more important limitation than weight.

Several unique features in the design and operation of the Soviet FMR reactor are of interest. The top lid of the pressure vessel is flat, whereas in USA designs it is hemispherical or semi-elliptical in shape. The control of the reactor is accomplished by replacement of the fuel rods by absorbing material through movement of some of the casings containing the fuel rods (~91 fuel rods per casing; 349 total number of casings in the reactor). In current USA practice used in power reactor design, the so-called fuel casings are spaced to have water gaps between them, within which absorbing blades (generally cruciform in shape) are moved in and out of the reactor for control. In some USA research reactors, and in one small power reactor in the USA (Army Package Power Reactor), the replacement of fuel by absorber is used for control.

The core flow passages will be orificed to distribute coolant flow to the casings according to their power output, thereby optimizing plant pumping power. These orifices apparently are adjustable and can be reset after actual starting of the reactor and determination of the neutron flux distribution. It is not explained how this is accomplished, but the advantage is great in that it eases the difficult problem of matching flow, through multiple orifices, with neutron flux distribution, both of which are difficult to predict closely. It would also make possible redistribution of flow to follow the shift in power generation through the life of the core. Orifices are used extensively in American power reactors, but none are adjustable.

For unloading of the casings (fuel elements) from the reactor, provisions are made for individual removal of each casing and also for unloading the core as a single unit. No details are given on the procedure followed in unloading the whole core. For unloading of the individual casings, the transfer of the spent casings from core to coffin is performed under a protective layer of water up to 15 feet deep; the coffins are stored under water in a holding tank located adjacent to the reactor pit. Cooling of the spent core in the holding tank is accomplished by circulation of the water in the holding tank through heat exchangers. Most of the operations involved appear to be performed by remote control.

As in the USA, the main circulating pumps in the high-pressure water loop are of the canned-rotor (gasketless) type.
In the Soviet PWR design, the initial charge of demineralized water in the high-pressure circuit is maintained at a high purity through the use of multiple evaporators. Water from the primary circuit is continuously drawn off in a bypass circuit where it undergoes evaporation in a three-stage evaporator. The distillate is returned to the primary circuit and the residue from the evaporators run off for eventual disposal. In the USA, water purity is generally maintained by means of ion-exchange beds rather than evaporators. The Soviets mention the possible use of ion-exchange columns, but discard them in favor of the evaporators because they believe the evaporators to be more dependable.

In contrast with current USA practice, no containment is provided in the Soviet PWR design to take care of the possibility of a reactor power excursion with consequent rupture of the reactor pressure vessel and release of fission products to the outside. The power stations are to be located 20 to 25 miles from large towns, and a 2-mile belt of housing clearance will be established around the power station. It is apparent that early atomic power stations in the USSR are to be located much further away from population centers than in the USA.

In the handling of radioactive liquid wastes under normal operating conditions, the Soviet requirements appear to be set up on the principle that no reactor system water or cleanup water be discharged from the plant into the ground or river. Instead, all plant water which could be contaminated for one reason or another, except for shower wash water which is properly purified prior to emptying into the sewer system, is cleaned up through the use of evaporative installations and returned to the plant. The residue from the evaporative process is run off into concrete tanks for extended storage. It appears that the waste disposal requirements in the USA are not as strict as the stated Soviet requirements. In the USA, slightly contaminated (radioactively "warm" as contrasted to "hot") water is often run into rivers; dilution prior to discharge and further dilution with river water is relied on to keep radioactivity below conservatively safe tolerances.

In general, the plant layout and arrangement of equipment is quite similar to that in USA practice. Health control procedures and safety practices are comparable to those in the USA.

The following table compares the fuel loading and fuel utilization attained in the operation of the USSR pressurized-water reactor as compared with a USA high-power reactor.
### TABLE V

**FUEL LOADING AND FUEL UTILIZATION IN A USSR REACTOR AS COMPARED TO A U.S. REACTOR**

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Fuel Loading</th>
<th>Lifetime Burnup, MWD/ton of Uranium</th>
<th>Lifetime (hr)</th>
<th>Conversion Ratio</th>
<th>Pu Obtained Burnt per Cycle</th>
<th>Pu-235 Burnt per Cycle</th>
</tr>
</thead>
<tbody>
<tr>
<td>Russian PWR</td>
<td>17 tons Nat.U</td>
<td>3,200</td>
<td>4,400</td>
<td>.80</td>
<td>115 kg</td>
<td>148 kg</td>
</tr>
<tr>
<td></td>
<td>23 tons 1.5%</td>
<td>Enriched U</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Steel-Jacketed</td>
<td>27 tons 2.6%</td>
<td>7,500</td>
<td>10,000</td>
<td>.714</td>
<td>121 kg</td>
<td>172 kg</td>
</tr>
<tr>
<td>Design*</td>
<td>Enriched U</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* Reputed to be the fuel conditions for the Yankee Power Plant design.

The foregoing data are, of course, based mainly on theoretical calculations so that there is no assurance at this time that the design conditions will be achieved. In addition to the nuclear considerations, the fuel burnups obtainable in practice will be limited by very practical considerations of irradiation damage to the fuel and corrosion of the fuel cladding. Experimental work in the USA indicates that the fuel elements of the above two designs should easily achieve burnups of the order of 3,000 to 5,000 megawatt-days/ton under the reactor environmental conditions without serious failures of the elements. At the same time, the experimental evidence is that burnups of 10,000 megawatt-days/ton can be achieved in the near future. The comparison in the foregoing table illustrates the basic differences in mode of approach to economically competitive nuclear power taken by the Soviets and the USA in their PWR designs. In the USA, the "first generation" nuclear power plants are being designed to achieve relatively greater fuel burnup and hence lower fuel reprocessing and fuel element fabrication costs. In contrast, the Soviet PWR design emphasizes low enrichment and high conversion ratio, while attempting only a relatively low fuel burnup (3,500 MWD/ton). The shorter fuel lifetime could mean a desire to produce plutonium, poor success with UO₂ fuel elements to date, or simply greater conservatism in stating the probable irradiation lifetime of the fuel. In this connection, it must be remembered that reactor designers in the United States are in the position of having to "sell" their reactor designs, and are not always modest in their projections of possible fuel performance. The Soviets state that various fuel charges will be experimented with in the pressurized water reactor, and mention particularly the use of more highly enriched fuel along with the natural uranium. This probably indicates that one of the plutonium recycle methods under consideration is that which utilizes "spikes" of plutonium diluted by non-fertile material. The use of the large pressure vessel for the pressurized water reactor may be of particular advantage in exploring various possible fuel charges.
No mention is made in the Belgrade report of the method of fabricating the zirconium-clad UO₂ fuel rods. The fuel rods are of about the same size as used in some USA designs and would be subjected to about the same temperatures, heat loads, corrosion conditions, and irradiation fluxes. Hence, fabrication techniques similar to those used in the USA for similar type fuel elements might be used. Cheap fabrication is important for obtaining economic nuclear power and is still a major obstacle to economic nuclear power in the USA. It is not known what progress the Soviets are making in this direction.

The pressurized-water type reactor, because of its relative compactness for a given power rating and its adaptability to supplying energy to steam-turbine power units, is suitable for propulsion applications. The power plant for the first USA atomic submarine is a pressurized water reactor and the majority of atomic submarines in use and under construction are powered by pressurized water reactors. Propulsion of surface naval vessels and merchant ships by pressurized water reactor power plants is both technically feasible and practical. Of the various reactor types possible, the pressurized water reactor is one of the better possibilities for application to locomotives and large mobile land units. The use of pressurized water reactors for aircraft propulsion is not feasible.

B. The Water-Cooled, Graphite-Moderated Reactor (APS)

The prototype of the water-cooled, graphite-moderated reactor is the reactor of the first atomic power station of the USSR, generally referred to as the APS Reactor, and described at the 1955 Geneva Conference. At the time of the Conference, it was stated that a 100 eMW reactor of this type was being built and would go into operation the following year (1956). That reactor has never materialized, but a reactor of the same type but higher output (200 eMW S-1-0) was described as part of the sixth five-year program (see Table I). A Soviet announcement was reported in reference U-5 of the start of construction on this latter plant. No further mention of the plant has been noted.

General Description

No description of the 400 MW plant has been given other than the performance specifications recorded in Table II. It is therefore necessary to infer its characteristics from those specifications and from the description of the APS Reactor. The APS Reactor will therefore be described briefly. A drawing of the reactor is given in Figure 6.

The permanent part of the APS Reactor proper consists mainly of a cylindrical assembly of graphite blocks pierced by axial holes which are referred to as fuel channels. The graphite assembly contains also channels for control rods, and is surrounded by a radiation shield. In the fuel channels are installed removable fuel assemblies, each of which consists of four steel fuel tubes surrounding a central feeder tube, all embedded in a long graphite cylinder. Coolant ($\text{H}_2\text{O}$) flows downward through the feeder tube and up through the four fuel tubes in parallel. Each fuel tube consists of two cc-axial steel tubes, with the uranium fuel occupying
FIGURE 6—5 MW POWER REACTOR

Any issues with the diagram can be discussed here.
the annular space between the two tubes. The inner steel tube is the pressure tube. It is 9 mm O.D. and has a wall thickness of 0.4 mm. The outer tube is 13.4 mm O.D. and has a wall thickness of 0.2 mm. Its purpose is to contain the fission fragments from the fuel, and probably to support the fuel. It has been stated that the "plasticity" (creep) of the uranium plays an important part in the success of the fuel arrangement, and it has also been stated that there is no metallurgical bond between the uranium and the steel tubes. It is not clear how a sufficiently good heat transfer contact is obtained and maintained between the fuel and the pressure tube.

An important characteristic of the APS Reactor is that the entire fuel assembly, including the pressure tubes, is replaced when refueling occurs. Consequently, the assembly must be relatively inexpensive if the reactor is to show favorable economic performance.

Some of the details of the APS design are summarized in Table VI.

In many respects, the APS reactor demonstrates a clever and original approach to the problem of designing a nuclear power plant which requires minimum research and development. In some respects, the performance of the reactor leaves a good deal to be desired, but it is not evident that the shortcomings are inherent in the reactor type but rather it appears that they represent compromises which were made in order to achieve the goal of power generation in a short time. It seems quite possible that rather large improvements in performance might show up in a later, larger version of the reactor.

Improvements over the APS reactor could conceivably be of two types: improvements in neutron utilization and improvements in thermal performance. An appreciable improvement in neutron utilization would occur automatically because of a size increase, which would reduce the loss of neutrons by leakage. This improvement could be utilized either to decrease the required enrichment of the reactor or to increase the conversion ratio. It is probable that the design would be made in such a way as to achieve some improvements in both of these directions.

It was stated by Blokhintsev that the increase of reactor size would allow the enrichment to be decreased to perhaps 2.5% and the conversion ratio to be increased to 0.5. It was not clear whether these estimates applied to a reactor using the same type of steel fuel tubes as the APS reactor, or whether they postulated a shift to a low cross section material such as zirconium. They appear to be optimistic for a reactor with steel tubes (Appendix III). Calculation indicates that a conversion ratio of 0.5 or a little better might be attained in a large reactor if the enrichment were maintained at 5%, and the lattice geometry were adjusted to capture the excess neutrons in \( ^{238} \text{U} \) resonances. Alternately, the large reactor might be designed to operate at an enrichment of about 3%, with a conversion ratio of about 0.32. Replacement of the steel by zirconium in a large reactor might allow a conversion ratio between 0.6 and 0.7 with an enrichment of 2.5%. Further increases in neutron economy might be effected if major changes in the geometry of the fuel elements were possible.
TABLE VI
CHARACTERISTICS OF SOVIET 5-MW ATOMIC POWER STATION REACTOR
(from Nuclear Engineering, November 1956)

| TYPE: | Thermal heterogeneous. |
| PURPOSE: | Experimental power production. |
| LOCATION: | Academy of Sciences, near Moscow. |
| CAPACITY: | 5 MW electricity from one turbo-generator. Heat rating: 30 MW. Four pairs of heat exchangers provided---one as stand-by. Maximum flux: 5 x 10^13 n/cm²-sec. |
| BURN-UP: | |
| CONVERSION FACTOR: | U-Pu: 0.32. |
| MODERATOR: | Graphite and light water. |
| CORE: | Over-all size: 5 ft. diam. x 5 ft. 6 in. high. Graphite, in steel tank, N₂ or Ne pressurized. Over-all size: 9 ft. diam. x 10 ft. high (approx.) 5 ft. above, 2 ft. below (approx.). |
| REFLECTOR: | |
| COOLANT: | Light water at 100 atm. Flows down centre of elements and up fuel tubes. Throughput at peak power: 300 tons/hr. Inlet temperature: 150°C. Outlet temperature: 270°C. Maximum activity: 0.2 c/l. After 1 minute: 2 x 10⁻⁵ c/l. |
| HEAT EXCHANGERS: | Maximum throughput: 40 tons/hr. Pressure: 180 psi. Temperature: 260°C. |
| CONTROL AND ROD WORTH: | Boron carbide rods, water-cooled. 6 near core center; worth 1.3% each. 12 near extremities; worth 0.7% each. 2 safety rods in core; worth 1.8%. 4 servo-controlled shim rods in reflector. |
| EXCESS REACTIVITY: | Temperature 2.5% F.P. Poisons 4.0% Burnup 4.5% Total 11.0% |
| SHIELDING: | Water: 40 in. radial thickness. Concrete: 120 in. radial thickness. |

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If the fundamental fuel tube design of the APS Reactor is to be retained---i.e., if the fuel tube must be discarded with the used fuel---it appears questionable that a shift to zirconium, which is expensive as a material and expensive to fabricate, would represent an economic improvement. It seems probable that any shift to zirconium for pressure tubes would involve design changes based on making the pressure tubes permanent or semi-permanent parts of the reactor. Such a design would be more closely related to that of the Hanford production reactors. A design change of this type would probably require a good deal of development work on the mechanical design of the reactor as well as on the production of suitable pressure tubes.

It would appear from the announced specifications of the 400 MW reactor that large improvements are being made in the thermal performance of the reactor. According to reference S-1-8, the outlet coolant temperature will be 540°C, a value which will allow the generation of steam superheated to 500°C. Reference S-1-9 quotes the coolant outlet temperature as 300°C, but states that the secondary steam is superheated steam at a temperature of 500°C. If these latter statements are true, one would have to assume that chemical superheating of the secondary steam is being used. However, the discussion of the reactor in reference S-1-8 seems to leave little doubt that the higher statement of the reactor coolant temperature was not an error: "These reactors will be graphite moderated, and the heat transfer media will be water and steam circulating through channels in the reactor. In some of these channels the steam in the first* circuit will be superheated to approximately 540°C. The heat taken from the reactor by the steam will be used in the steam generator to produce and superheat secondary steam..." If this performance is achieved, the reactor will be a very attractive one from the thermal point of view. The reactor apparently would operate as a forced circulation boiling reactor, with steam generation in some of the fuel tubes and recirculation of the generated steam through other tubes to superheat it.

A number of complex engineering problems would be involved in such a design, but the principle would appear to be fundamentally feasible. It would be surprising if zirconium were used in a reactor of this type because there is question as to the performance of zirconium in very high temperature water. It would seem more probable that the actual reactor is retaining the original steel construction and is minimizing the effects of the relatively poor nuclear performance by increasing the thermal efficiency, and perhaps by other means. It is not self-evident that the neutron utilization with steel is so poor as to constitute a fatal deficiency of the reactor. There is some possibility that the nuclear performance might be improved by using a blanket of natural uranium fuel for part or all of the superheating. Other opportunities for improvements may exist which are not apparent until the reactor type is studied in detail.

* The "first circuit" probably means the primary, or reactor coolant, circuit.
SECRET

A further possibility for this reactor type would be direct utilization of reactor steam in the turbine, as is done in the USA direct-cycle boiling reactors.

Comparison with USA Technology

There is no reactor under construction or development in the United States which is strictly comparable with the water-cooled, graphite-moderated power reactor. The Hanford reactors are, of course, of this type, but the design considerations for a power reactor are rather different than for a production reactor.* Pressurized tube type power reactors have been receiving much attention in the United States in the past year; the pressurized tube types which are being actively pushed are all moderated by D_2O. There are certain advantages and disadvantages of the pressurized tube concept which have varying degrees of importance depending on the moderator and coolant used, but they do apply to some degree to all pressurized tube reactors. These are the following:

Advantages

1. The pressurized tube reactor is not limited in size by the necessity for construction of a pressure vessel.

2. Extrapolation of design from a small reactor to a large one involves less effort and less uncertainty. The increase in size of such a reactor is accomplished mainly by the addition of more pressure tubes which need not differ greatly from those used in the smaller reactor.

3. Refueling, and particularly partial refueling, of the pressurized tube reactor is usually easier. This characteristic makes feasible the shifting of fuel elements from one reactor position to another in order to maximize fuel lifetime and/or to flatten the power distribution.

4. The elimination of a large pressure vessel reduces the magnitude of possible system failures, and in this respect increases safety.

Disadvantages

1. The necessity for containing high pressures by tubes which are located in regions of high neutron flux inevitably increases the parasitic absorption of neutrons. The use of low cross-section materials for pressure tubes in order to minimize the parasitic absorption can prove to be quite expensive.

2. The provision of headers and external piping to feed all the pressure tubes can lead to a complex design. Space restrictions imposed by

* Experimental and developmental work has been done at Hanford, aimed toward power reactors patterned after the Hanford reactors, but employing zirconium pressure tubes.
header and flange problems can force the design away from optimum performance in certain cases.

It is not possible to make any general statement as to the relative costliness of pressurized tube versus pressure vessel construction. It is probable that each type has its own area of superiority.

With respect to its coolant system, the APS type reactor is actually a variation of the pressurized water type. The coolant performance of the APS reactor exhibits the following characteristics:

1. The heat transfer performance is high: Higher heat fluxes are used than in USA practice for pressurized water reactors. The average heat flux in the APS reactor is stated to be $5.5 \times 10^5$ Btu/ft²hr; this is close to the maximum value in USA PWR design. Local boiling is allowed in the APS fuel channels. This has not generally been the case in USA designs. Bulk boiling and steam cooling are apparently being used in the power station No. 2. The temperature rise of the coolant in going through the core of the APS reactor (approximately 135 °F) is several times the value used in USA practice. It is probable that the above characteristics are to a large extent the result of the pressurized tube design, and of the fact that the fuel elements are relatively long. It can be said, however, that the technology, with respect to heat removal by water, is advanced.

2. Water purity is maintained by distillation rather than by ion exchange as in USA practice. It is surprising that this same scheme is retained even in the large pressurized water reactors of station No. 1.

With respect to fuel element technology, the statements of Blokhintsev would indicate that the APS element design is primarily a mechanical rather than a metallurgical one. Without more information on the characteristics of the fuel element, it is not possible to determine to what extent the design restrictions on the element represent a limit in the attainment of good nuclear performance. It is possible, since the success of the element depends upon the "plasticity" of the uranium, that there may be rather severe restrictions on the usable ratio of uranium to jacket material. However, the general approach of using mechanical contact rather than a metallurgical bond between fuel element and jacket may be a good one. It is the approach used by the British in their gas-cooled reactors, and is used in many of the current USA water designs, although in the latter cases the fuel material is uranium oxide rather than metal.

Method of Operation

The APS reactor is designed to have negative reactivity coefficients with respect to coolant density and coolant temperature. Its dynamic characteristics therefore are probably much like those of a pressurized water reactor. The reactor will tend to match load automatically, and would not be expected to exhibit any unusual control problems. The reactors for power station No. 2 will probably have similar characteristics, although they may be modified considerably by the boiling of the coolant. Design
for a negative coolant density reactivity coefficient implies that if coolant escapes into the moderator because of the failure of a pressure tube, the reactivity will be increased. This characteristic was possessed by the APS reactor, but the designers were able to convince themselves that the total possible reactivity increase was not dangerous. This characteristic would have caused some concern in the United States. Presumably, the reactors of power station No. 2 will have the same characteristic qualitatively, but its magnitude will depend upon the specific design. With respect to safety instrumentation and general safety practice, the APS reactor and its operating procedures appear to be comparable with USA practice. A major difference in design practice with respect to reactor safety is the absence of a gas-tight containment structure. This is characteristic of the Soviet approach. Their statement is that they rely upon isolation of the site rather than upon containment structures.

Reloading of the APS reactor is accomplished by using the reactor room as a large hot cell. The remote operations required are simple ones and can be carried out by means of a crane operated remotely by direct vision through a shielding window. This method has not been used in the United States for power reactors. USA methods usually manipulate fuel elements under deep pools of shielding water or employ coffins. The APS reactor allowed complete flexibility for repositioning fuel assemblies, but does not allow segmental reloading of fuel in the individual channels.

Fuel Cycle

It may be expected that the enrichment of the reactor will be relatively high—probably not less than 3% if steel is retained as the fuel tube material. Under these conditions, the conversion ratio will be low. It will probably not exceed 0.5 in the steel reactor, even if a considerably higher fuel enrichment is used. If zirconium pressure tubes are used and if a fuel element design is used which allows rather large fuel sections, the enrichment can be quite considerably lower, and the conversion ratio may be relatively high. If the reactor type were developed in the United States, it is probable that a design optimized for best economies would not have a conversion ratio higher than about 0.7, even with zirconium tubes. In the USSR, it is probable that any shift toward the use of zirconium will be accompanied by an attempt to improve the neutron economy sufficiently to allow operation with natural uranium feed and plutonium re-cycle. It is not obvious that this goal can be achieved with a reactor of this type which is economically attractive in other respects. Of course, it is very improbable that the goal could be achieved with a reactor using steel pressure tubes.

The fabrication of fuel tubes of the type used in the APS reactor is probably straightforward. It is almost certainly a routine procedure by this time. The tubes are more complex than fuel elements which do not serve also as pressure tubes, but the use of steel may keep them from being extremely expensive to manufacture. Nothing can be said about the possible costs of other types of fuel elements which might be used in the Power Station No. 2 in order to improve neutron economy.
Propulsion Applications

The water-cooled graphite-moderated reactor is not the most compact reactor type. As exemplified by the AEC reactor, however, it does have a reasonably high power density, and when allowance is made for the greater flexibility available in location of control rods and for the advantages of reloading by individual fuel channels, it may be a serious competitor with the pressurized water reactor for the propulsion of sizeable marine vessels. The diameter of the reactor core required for a 200 MWe reactor (the capacity of the icebreaker reactor) would probably be between 9 and 10 feet (Appendix III). Such a diameter would not be out of reason for the application, particularly if the good steam conditions claimed for power station No. 2 could be attained. The reactor is probably inferior to the pressurized water type for compact naval vessels such as submarines. It is probably not suitable for other propulsion applications.

C. The Gas-Cooled, Heavy-Water Reactor

The gas-cooled, heavy-water reactor was mentioned as a type for one of the major power stations in the early statements of the Five-Year Plan. It has not been mentioned in later descriptions of the current program. The available description of such a reactor is one which was given in a paper by V. V. Vladimirov at the Fourth Annual Conference on Atomic Energy in Industry in New York in 1955. If the reactor is still receiving serious consideration, its design may now be quite different from that which was described in this paper. Therefore, the comments below must be interpreted as referring to Soviet design as it existed in 1955.

Description of Reactor

The gas-cooled reactor is a calandria type with a gas inlet temperature of 90 °C and an outlet temperature of 420 °C. The coolant is used at a pressure of 40 to 60 atmospheres; the pressure-vessel diameter is not given. The fuel elements are thin rods or plates of natural uranium, clad with an unspecified light metal which will withstand a maximum temperature of 550 °C. The coolant is CO₂ at 40 to 60 atmospheres, and steam is to be generated at 400 °C and at two pressures: 29 atmospheres and 2 atmospheres. The quite low specific D₂O inventory of 0.2 to 0.4 tons per megawatt of electrical power is claimed. It appears that this low inventory is achieved at the price of expensive reactor design and some loss of reactivity lifetime. It would appear to indicate that D₂O is quite expensive in the Soviet Union.

Comparison with USA Technology

The gas-cooled, D₂O reactors which have been proposed in the United States have been of the pressure-tube rather than the pressure-vessel type. The general advantages of pressure-tube reactors have been discussed in section V-B above. It is probably true, however, that for an electrical output in the 100 MW range, a higher ratio of power to D₂O inventory can be attained with the vessel type reactor. It is apparently this consideration which influenced the Soviet design in 1955 and at earlier dates.
The major differences between the Soviet design and the design practice that would probably be followed in the United States for a reactor of the same general type apparently stem from their desire to limit the D_2O inventory. The power density is quite high for a gas-cooled reactor, the power required for the circulating blowers is quite high, and the inlet gas temperature is low (90°C) in order to obtain a high temperature rise in the reactor and therefore a high energy output per unit of gas circulated. The design description does not contain enough detail to allow an evaluation of the effects of these design principles on other aspects of the reactor performance. One would guess that the nuclear performance has suffered as a result of the emphasis on power density. The description claims that the reactor will operate on natural uranium; such operation, if possible, must be marginal. Insofar as one can judge from the rather meager description of the reactor, the major differences from USA practice arise from differences in design choices rather than from differences in reactor technology. It does not seem possible to make important inferences from the description other than perhaps that the D_2O supply was limited at the time of the design study. It does appear that in this particular reactor type the Soviets have altered their usual approach, which appears to be to aim their developmental program in the direction of reactors which avoid as far as possible any limitations with respect to ultimate power output. It may possibly be that this reactor is not mentioned in current announcements of the reactor program because its fundamental design is being altered through research and development.

Operation

No description of operating procedures for the reactor has been given. It would appear that the reactor cannot conveniently utilize segmental loading of the fuel, one of the principal operating advantages of the pressure-tube type D_2O reactors being considered in the United States.

Fuel Cycle

It is quite feasible to design a D_2O-moderated gas-cooled reactor, either vessel type or tube type, which will operate on natural uranium. It is very probable that the Soviets would strongly favor such a design, and it can probably be assumed that if the reactor is built it will either operate on natural uranium or will be so designed that natural uranium operation can be reasonably expected in the full-scale version. This implies certain design restrictions which are discussed below. If these restrictions should prove too much of a limitation on economic operation, the design could conceivably shift to a slightly-enriched one with the aim of supplying the enrichment ultimately by plutonium recycle. Such a change in objective would probably be made only after very serious consideration.

Possible Status of the Reactor

If natural uranium operation is to be achieved, and if the power density in the reactor is to be reasonably high, there appear to be only three possible materials for fuel element jackets: aluminum, magnesium
and beryllium. The first two of these materials limit the maximum coolant temperature to about 800°F, or less. With beryllium jackets, attractively high temperatures can be attained, particularly if the fuel is in the oxide form. The development of beryllium jackets which are reliable and economically feasible is probably a quite difficult task. It is one which the British are evidently working on and which is also of interest to the United States. It seems quite probable that the lack of this type of fuel jacket, possibly coupled with some shortage of D₂O, has caused a decision to postpone this reactor type until research and development can proceed further. There are a number of considerations which make this reactor type one which cannot be adequately proved in a small size pilot plant. It would therefore seem quite reasonable that research and development would be continued until the status of the development is such as to justify a reactor of relatively large size.

D. Boiling Water Reactor

One of the four experimental power reactors to be constructed and operated as part of the USSR Sixth Five-Year Plan (1956-1960) is of the boiling water type, S-1-8, S-1-9. The plant, described by the Russians as a relatively small capacity plant, is to produce about 300 megawatts of heat and generate 70 megawatts of electricity. The Soviets state that this station, being considered experimental, will require thorough experimental study before it can be recommended for use in large power stations. It is interesting to note the distinction made by the Soviets between the experimental plants and the large industrial power stations. Basically, both categories of plants are experimental and exploratory in nature. For the experimental plant, the principal objective is to establish the technical problems involved and to work toward their solution; for the industrial stations, the technical feasibility has presumably been established and the prime objective is to explore various approaches toward the attainment of economically competitive nuclear power.

Little detailed information has been released on this reactor project. Plain water is used as the moderator and heat-transfer medium. The saturated steam generated in the reactor at a pressure of 30 atmospheres will run a 70 megawatt turbine. The steam-water mixture in the reactor will have natural circulation flow. In this respect, the Soviet design differs markedly from USA plans in that the USA uses forced circulation of the water coolant rather than natural convection for boiling reactors of the size specified in the USSR design.

The seriousness of the problem associated with prolonged operation of the turbine on radioactive steam will be checked. The Russians are evidently prepared to do maintenance work on the turbine equipment by remote control. It is planned to obtain data on reactor stability as affected by steam content in the reactor and on the regulation of the entire power plant.

The Soviets also plan to operate the reactor using a binary cycle. In this cycle, in addition to the steam produced directly in the reactor, steam is also produced in a secondary water circuit from the heat of that
portion of the water flowing through the reactor but not changed to steam; the heat transfer from the reactor water to the fluid in the secondary circuit occurs in a steam generator. The Russians state that this second source of heat will produce steam in an amount up to 1/3 of the steam produced directly in the reactor.

The boiling-water type of reactor was pioneered by the United States, at the Argonne National Laboratory. The results of this pioneering work were disclosed to the world in a paper U-25 presented on July 18, 1955, at the International Conference on the Peaceful Uses of Atomic Energy in Geneva, Switzerland. The Soviets showed considerable interest at that time in the fact that direct experiments had been run which proved the technical feasibility of a boiling-water reactor to generate electricity. It was obvious that, prior to the presentation of the paper, the Soviets had grave doubts as to the possibilities of safely producing steam directly in the reactor and hence had not pursued the development of a boiling-water reactor.

On the basis of the announcements in the Belgrade report, the Soviet Union's boiling-water reactor is being built to explore the technical aspects rather than the economic considerations. They are evidently still in the stage of learning about the behavior of such a reactor and about the special problems of power-plant operation incurred by the use of the boiling-water reactor. In this respect, the Soviets are undoubtedly lagging behind the USA due to the initial USA lead in boiling-water reactors.

It is significant to note the relatively large-size reactor unit the USSR plans to use to study the technical problems involved. The EWR U-30 recently constructed and operated in this country for experimental purposes, produces 5 MW of electricity; in decided contrast, their experimental unit is designed for 70 electrical megawatts, and uses natural circulation of the coolant.

The size of the Soviet experimental boiling-water reactor corresponds more closely to the USA first-generation nuclear power plants being built to develop the economic feasibility of nuclear power. The Dresden Nuclear Power Station U-29 under construction near Chicago is of the boiling-water type and utilizes a dual (binary) cycle. The electrical output is to be 180 MW and top steam pressure 1000 psi (68 atm.). The secondary to primary steam ratio is to be of the order of 0.8 compared with a value of 1/3 quoted by the Soviets for their reactor.

The boiling-water type of reactor is adaptable for propulsion application in much the same way as the pressurized water reactor. Where compactness of reactor unit rather than economics is of prime importance, as is the case in military applications such as naval vessels, the pressurized-water reactor (perhaps slightly boiling) may have inherent advantages over the boiling-water reactor. However, in commercial applications wherein cost considerations are important, reactor designs may be optimized in the direction of producing fissionable from fertile materials as well as
producing motive power. Hence, reactor unit sizes will increase so that boiling-water reactors may possess certain advantages over the pressurized water reactor. There are certain limitations in the design of boiling-water reactors due to the pitching, rolling, and accelerations of marine vessels in stormy water which can adversely affect the operational stability of the reactor unit. It is probable that forced circulation must be used in boiling reactors for such applications. It is believed that boiling-water reactors will eventually have a place in the propulsion of commercial marine vessels, although they are not feasible for aircraft propulsion.

Since no indications have been given as to the progress of the experimental boiling water plant, it is worthwhile to speculate briefly on its possible status. If it is assumed that the technology of pressurized water reactors has developed to the point that construction of power plant No. 1 is indeed under way, then it seems quite probable that there are no unsolved feasibility problems impeding the construction of some kind of a boiling reactor. It is quite possible that the boiling water plant may be the first of the experimental plants to operate. It is true, however, that the proposed output is quite high for a natural circulation boiling reactor, and that the building of such a plant will be a rather large-scale undertaking. The pressure vessel, for example, may be as large as that for the pressurized water reactor, although its design pressure will be lower. Delays could arise through reconsideration of the basic design goals of the reactor. Since this reactor will be the first boiling reactor built by the Soviets, it may be expected that much work will go into its planning in an attempt to make it the biggest possible first step toward the ultimate goal. It is to be anticipated that a rather large amount of physics experimentation will be required also for the design of the reactor.

E. Homogeneous Reactor

The Soviet Union's version of the homogeneous reactor, according to their reply to the UN questionnaire, S-1-9 is to be a boiling water, natural circulation type, with heavy water moderation. This reactor will probably begin operation with a fully-enriched U\textsuperscript{235}-thorium fuel (oxide slurry), with plans for later operation with U\textsuperscript{233}-thorium. The basic design features are most likely still those outlined by Alikhanov, et al., at the Geneva Conference. S-3-7 It is not known whether the reactor will be of the single-region type with an intimate mixture of U\textsuperscript{235}-Th or of the two-region type, with the uranium fuel in the core surrounded by a thorium blanket. Due to the technical difficulty of fabricating the core tank, and the uncertain advantages of a two-region reactor, the single-region type may be preferred. This does not imply that the Soviets would not be competent to fabricate the core tank, but rather that their interest (as evidenced by the fact that this type is the smallest of their planned experimental reactors) and the high cost of the core tank fabrication would probably lead them to prefer the simpler single-region approach. This is similar to the type of reactor (PAR) now being considered in the USA by the Pennsylvania Power & Light Company. U-1
The Soviet version of the homogeneous reactor differs from USA design in that they have proposed a boiling reactor, whereas the USA plans call for a circulating fluid fuel. The advantage of a boiling reactor lies in the reduced fuel and D_2O inventory, since the fuel remains in the reactor and additional fuel solution is not required to circulate in piping outside the reactor itself. However, a boiling homogeneous reactor is limited, by the rate of heat removal at reasonable steam voids, to specific powers of the order of 20 kw/l, and the published information on the Russian homogeneous reactor indicates that its specific power will be about 13-20 kw/l. A reactor of this type, while potentially suitable for large marine or railroad propulsion, does not operate with a sufficiently high specific power or at sufficiently high temperature to be suitable for aircraft propulsion.

In many, if not most, propulsion applications, the requirement is for a compact reactor of high power density and small physical dimensions. This requirement is not compatible with the conversion of fertile material unless the total power requirement is quite high. In the cases where conversion must be quite low because of the restrictions on reactor size, the homogeneous reactor may ultimately be an attractive possibility for propulsion because it can achieve reasonably low fuel cost without conversion. At the present stage of homogeneous reactor technology, in the United States at least, it is not an attractive possibility for propulsion.

A recent Soviet announcement, U-40, from Radio Moscow, told of plans to operate a locomotive by nuclear power. The reactor for this locomotive, described as an "uranium salt solution, and coupled to a gas or steam turbine," could conceivably be the experimental homogeneous reactor station.

Future Soviet developments will probably be toward increasing the specific power of the reactor by using forced circulation. The next five-year plan will probably include a forced circulation experimental reactor, and possibly a larger version of the presently scheduled boiling homogeneous reactor.

The available information on the status of the homogeneous reactor is extremely limited. The one technical summary of the work, which was given at Geneva, gave the impression that most of the investigation is still in the laboratory stage. The problems were discussed in a quite general way, and statements regarding the development of certain components, such as full flow recombiners for dissociated water and devices for reducing radioactivity in the steam, were so optimistic as to indicate either that the Soviet technology was far ahead of that in the United States, or that it had not yet advanced sufficiently to have revealed the difficulty of the practical problems. In view of the very extensive effort that has been devoted to aqueous homogeneous reactors in the United States, it is improbable that the Russian technology was more advanced in 1955, or indeed that it is so today. It is doubtful if a large-scale power station, using the homogeneous principle, will be in operation in the USSR until...
nearly 1970, and probably not until after 1970, although future developments, either in the USA or in the USSR, could induce the Soviets to construct one or more large-scale homogeneous power plants during their 6th five-year program (1965-1970).

The potential economic advantage of homogeneous reactors lies in the low fuel fabrication costs, the potentially lower fuel processing costs, and the higher available conversion ratio. These are counterbalanced by the requirement of expensive heavy water and the high costs of containment and maintenance. A critical evaluation of a homogeneous type reactor U-2 (a single fluid two-region solution type non-boiling reactor) places the capital cost in the USA at $376/kw (probably an under-estimate) and estimates the electricity cost at 19.4 mills/kwh. There is no reason to believe that the Soviet version of the homogeneous reactor would have any significant advantage over similar designs in the USA.

F. Fast Reactor Program

The Soviet fast reactors are designed to operate exclusively with plutonium as a fuel. Plutonium is a much better nuclear fuel than U235. The number of neutrons produced per capture in the fuel for fast neutrons (~0.2 Mev) is approximately 2.64 for Pu and 2.20 for U235 so that a much higher breeding ratio is possible with Pu. However, the USA Fast reactors, EBR and APDA, are designed to operate on U235 fuel in order to allow the development of reactor technology to proceed without being hampered by the problems of plutonium technology. The Soviet plans to use plutonium exclusively are probably based on considerations of availability of U235 rather than considerations of reactor performance.

One fast reactor has been described in detail by the Soviets. This reactor, called the BR (bystryy reaktor - 2), used plutonium fuel rods, mercury as a coolant, and has a blanket of depleted uranium. Its heat output is 200 kilowatts. BR-1 was probably a fast reactor cold criticality experiment. The BR-2 has been used mainly to determine the physical constants of fast neutron power reactors. According to the Soviets, this reactor was to be rebuilt in 1957 to obtain a considerable power increase. S-2-32

Originally the development of nuclear power in the Sixth Five-Year Plan called for the construction of an experimental power station using a fast reactor. The fuel would be plutonium, the coolant sodium, and a blanket of U238; the electrical output was specified as 50 MW. S-2-33

Because the fast reactor represents the only obvious method for utilizing a large fraction of the energy potentially available in U238, it is a reactor type which is rather generally considered to have an important place in the nuclear-power field. Most countries which have serious nuclear power programs consider that they will eventually use the fast reactor as part of their program. Sufficient experience has been obtained with the fast reactor to indicate that it will not be the cheapest method of producing nuclear power in the immediate future. Furthermore, unless the country in question has a large supply of fissionable isotopes (either
separated U235 or Pu), the initial fueling of such reactors may present a problem. It is therefore to be expected that the development of fast reactors will proceed at a somewhat slower pace than that of thermal reactors, and there is no apparent reason why the situation will be different in the USSR. On the other hand, the goals to be met in the fast reactor technology are perhaps somewhat better defined than are those in the field of thermal reactors and the range of design variables is apparently smaller. It may therefore be that the fast reactor development, over a long period, will proceed in a more direct path toward its goals. The Soviet objective of building a 50 eMW plant by 1960 is an ambitious one, particularly in view of the intention to fuel with plutonium, and it would not be surprising if this reactor developed a lag in its schedule. It is improbable that fast reactors will account for a large fraction of the nuclear generating capacity in the USSR during the period from the present to 1970, although it is to be expected that development and construction of fast reactors will continue.

G. Sodium-Cooled, Graphite-Moderated Reactor

A reactor moderated by graphite, using molten sodium as a heat transfer medium, is scheduled for one of the experimental power plants. S-1-8 Sodium is being used to enable the plant to generate high-parameter steam to drive the turbines. The plant is to provide experience on the operation of the sodium cooling system, and at the same time to indicate the neutron physics characteristics of a graphite reactor using sodium as heat transfer medium. This reactor is presumably similar to our SRE (Sodium Reactor Experiment), which generates 20 MW of heat and is a pilot plant for a much larger sodium-graphite reactor, U-13

Although the sodium-graphite reactor is an important possible type because of its potentialities for high temperature operation, experience in the United States raises some question as to whether this advantage outweighs its disadvantages. The neutron economy of the reactor, at least in its U.S.A. form, is rather poor, and its cost appears to be high. It is quite possible that new developments could change both of these characteristics. However, such developments would have to be of a basic nature, not incremental improvements on existing technology. There is no indication that the Soviets have made such developments. It is doubtful that the reactor in its present form could operate with natural uranium feed and plutonium recycle. The British, who have also studied this question rather extensively, apparently have serious doubts as to the feasibility of natural uranium feed. Because of these limitations on nuclear performance, and the difficulties of the reactor type, it is one of the more likely candidates for a schedule lag if such lags develop in the Soviet program.

The basic reactor type represented by the Sodium Graphite Reactor is one which may have potentialities for propulsion use, both for marine craft and aircraft. For such applications, some other moderator might be substituted for graphite, or conceivably an unmoderated reactor could be used. In any case, an interest in liquid-metal-cooled reactors for military pro-
pulsion purposes could conceivably accelerate the development on both the Sodium Graphite Reactor and the Fast Reactor to a degree which is out of proportion to their importance in the central station power program.

H. Icebreaker Ship Reactor

The Soviet nuclear powered icebreaker, as scheduled for their Sixth Five-Year program, is to be a 440' long, 904' beam, 16,000 ton vessel with a 44,000 hp (33 MW) turbo-electric drive, U-14, S-1-5. This ship, the "Lenin," is being built in Leningrad and was scheduled to be launched November 7, 1957, but was actually launched on December 5, 1957. The power plant is located forward, and drives three turbo-electric generators, each of which powers one screw (3 screws total). A total of 40,000 hp will be used for propulsion.

No information on the type of reactor is available, other than it is a 200 MW reactor, and is expected to operate on one fuel loading for about two to three years. S-1-5. The size of the vessel is sufficient to house either the PWR (pressurized-water reactor) or the AGR-1 type (water-cooled, graphite-moderated) reactor. Full-time use of the reactor at full power would consume about 60 kg of U-235 annually as compared with about 150,000 tons of diesel fuel for an equivalent conventional vessel. Actually, the use factor for the vessel must be far below unity because of the nature of its application.

All but one of the USA ship propulsion reactors are of the PWR type. Nautilus experience U-15 has shown that submarine propulsion using a PWR type reactor is quite successful. This nuclear powered vessel has cruised with a single fuel loading for over two (2) years (69,000 miles total), which is comparable to the performance expected from the Soviet icebreaker.

VI. Chemical Processing

Very little information is available on Soviet processes for chemical separation of the valuable fissionable material from the irradiated fuels. Spent fuel elements, it is believed, are sent to a central station for chemical processing. The Soviets have obviously operated processing plants for a number of years in order to handle the product of their plutonium-producing reactors. Presumably, their scheme for chemical processing of irradiated fuel elements will be essentially the same as that used in this country. These processes would differ for the various types of fuel elements from the different reactors only in the "head-end" process necessary to dissolve the fuel element (see Table VII). Once the fuel element is dissolved, the balance of the chemical processing stages could be the same for the fuel from a number of different reactors. The principles of chemical processing for the separation and purification of plutonium, uranium, and thorium are the familiar reactions on which common analytical procedures are based. Considerable detail of USA processing methods are available in unclassified literature. (Summary in references U-3, U-4, and U-5.) These may include precipitation, ion exchange, solvent extraction, distillation, and electrolytic processes, any of which can conceivably be used successfully. The most practical separation processes in use in the
<table>
<thead>
<tr>
<th>Reactor</th>
<th>Fuel Element</th>
<th>Possible Process</th>
<th>Notes</th>
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<tbody>
<tr>
<td>(1) PWR</td>
<td>Zr clad, UO₂ fuel</td>
<td>Redox or Purex</td>
<td>Physical means may be used to separate UO₂ from cladding</td>
</tr>
<tr>
<td></td>
<td>Possibly ThO₂ later</td>
<td>Thorex</td>
<td></td>
</tr>
<tr>
<td>(2) Graphite-moderated</td>
<td>Stainless steel or Zr clad U metal</td>
<td>Redox or Purex</td>
<td>Sulfuric acid or hydrofluoric acid</td>
</tr>
<tr>
<td>(3) D₂O-moderated, Gas-cooled</td>
<td>Al or Be clad</td>
<td>Redox or Purex</td>
<td></td>
</tr>
<tr>
<td>(4) Na-cooled, Graphite</td>
<td>Probably Zr-Al alloy:S-1-2</td>
<td>Redox or Purex</td>
<td></td>
</tr>
<tr>
<td>(5) Na-cooled, Fast*</td>
<td>Uranium or Thorium blanket</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(6) Boiling water</td>
<td>Zr or stainless steel clad UO₂</td>
<td>Redox or Purex</td>
<td>Same as for graphite-moderated, water-cooled station</td>
</tr>
<tr>
<td>(7) Homogeneous</td>
<td>UO₂-ThO₂ slurry</td>
<td>Thorex</td>
<td>Probably on-site processing plant</td>
</tr>
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* In the United States, pyrometallurgical reprocessing is under active development for fast-reactor fuel. No reports have been noted of such work in the USSR. Its use probably requires that fuel elements be re-fabricated from rather highly radioactive material; hence, fuel elements must be of suitable design to make the process attractive.
USA for large-scale separation of irradiated fuel elements are the solvent extraction methods; Purex or Redox for separation and purification of plutonium and uranium, and Thorex for thorium and uranium processing. These processes all involve dissolution of the fuel element, and subsequent extraction from a nitric acid solution into an organic solvent—hexone for the Redox process and tri-n-butyl phosphate for the Purex and Thorex processes—leaving fission products and other impurities behind in the aqueous phase. In the Purex and Redox processes, plutonium is reduced to the trivalent state and removed by nitric acid solution, while in the Thorex process thorium is recovered by eluting into a dilute nitric acid solution. In all three processes, the uranium remaining in the organic phase may be recovered by eluting into a mildly acid water solution, and if desirable further purified by an ion-exchange process.

The principal characteristic defining the requirements of the head-end process is the nature of the fuel element cladding. In general, it is desirable to remove as much of the cladding as possible by physical means, such as cutting and stripping, before attempting to dissolve the remaining portions of the fuel element. The reagents used for fuel element dissolution are (1) nitric acid for aluminum metal, uranium metal, thorium metal, $\text{UO}_2$, ThO$_2$, and uranium-aluminum alloys; (2) sodium hydroxide for aluminum and uranium-aluminum alloys; (3) hydrofluoric acid for zirconium and zirconium alloys; and (4) sulfuric or hydrochloric acid for stainless steel or stainless steel matrix containing $\text{UO}_2$. For thorium and thorium oxide, a small amount of hydrofluoric acid in the nitric acid reagent is necessary to enhance the rate of dissolution.

In the USA, the cost of chemically reprocessing spent fuel elements has been tentatively established by the USAEU at $15,300 per ton of fuel. U-6 This, of course, does not take into account the cost of shipping, chemical cleanup, process losses, or waste disposal, which can account for increases up to a factor of two in some cases. As in all chemical processes, the specific cost is lowered by the use of plants of large capacity and is increased by any necessity for building diversity into the plants. In these respects, Soviet plants may operate at an advantage over USA plants, since the complete government control over types of reactors built and types of fuels used should minimize the flexibility required of a given plant and maximize the unit size of plant.

In addition to chemical processing of the irradiated fuel elements, it is necessary in all water-cooled reactors, to continuously purify the coolant water. The proposed scheme to be used on the FR reactor, S-1-9 and presumably the same as will be used on other water-cooled power reactors, is multiple distillation, similar to that used on the first Soviet power reactor, the APS-1. S-2-2 The water purification system is different from normal USA practice in that multiple evaporators, rather than ion exchangers and filters, are used. Although ion exchange purification was considered, it was rejected in favor of distillation methods as being more dependable, thus indicating that the present quality of Soviet ion exchange resins is inferior to similar products in the USA.
In the Soviet system, S-1-9 a certain quantity of the primary coolant stream is bypassed into a special purifier where it is evaporated in a three-stage evaporator, and then in dish evaporators. From the evaporators, the distilled water is pumped back into the primary reactor circuit, and the residue containing the radioactive material is sent to reinforced concrete dump tanks for storage. Apparently the possibility of combining the concentrated liquid waste with concrete for ultimate disposal is presently under investigation.

Waste Disposal

The question of safe disposal of radioactive waste appears to be a major problem in the USSR as indeed it is in every country planning any extensive use of nuclear power. The Soviet reply to the UN questionnaire S-1-9 indicated that they are attempting to combine the liquid waste (concentrated) with cement to preserve it in solid form. Krotkov, S-3-27 at the Geneva Conference, reported that the Soviets had studied the problem of disposal in natural waters, such as rivers or oceans. Their work indicated that a substantial fraction of the radioactivity was absorbed in the muds and bottom deposits, and in the living organisms in the water. Therefore, discharging radioactive waste to the oceans was not considered a safe and reliable method of waste disposal. In his talk, Krotkov said that "the problem of radioactive waste disposal cannot be solved at the level of any single country; it calls for sanitary control at the international level."

VII. Economics

The status of reactor development in the USSR, as in this country, is such that any of the proposed reactors are either technically feasible now, or may reasonably be expected to be so in the near future. The prototype reactors are experimental in the sense of "what is the best answer to the existing problems," rather than "can an answer be found." Since almost any of the reactor types may conceivably be built on a large commercial scale, the principal remaining question is concerned with economics. Since cost data in the USSR are not available, equivalent costs as they would exist in the USA have been considered here. It should be emphasized that the results obtained are quite general and should not be used as an index of actual costs of nuclear power in the USSR or as indicative of future trends, without due consideration of the economic structure and industrial development in the USSR. Of the two types of large-scale power plants proposed, sufficient information is available for only two, the PWR type and the APS type, to permit a cost estimation.

An estimate has been made of the fuel costs for the Soviet PWR power station based on USA standards. The results of the estimates are given for two fuel burnups, namely, (1) the expected burnup of 3,500 megawatt-days per ton of uranium, and (2) a burnup of 10,000 megawatt-days per ton, which the Soviets hope to achieve in the future. The various items determining the over-all costs are listed separately in Table VIII.
### TABLE VIII

**FUEL COSTS**

Soviet PWR Station No. 1

<table>
<thead>
<tr>
<th></th>
<th>3,500 MWD/ton</th>
<th>10,000 MWD/ton</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Use Charge</td>
<td>0.1</td>
<td>0.1</td>
</tr>
<tr>
<td>Fuel Fabrication</td>
<td>7.7 - 12.9</td>
<td>3.1 - 5.1</td>
</tr>
<tr>
<td>Fuel Reprocessing</td>
<td>1.2</td>
<td>0.4</td>
</tr>
<tr>
<td>Net Fuel Burnup</td>
<td>1.0</td>
<td>0.8</td>
</tr>
<tr>
<td><strong>Total Fuel Cost</strong></td>
<td>10.0 - 15.2</td>
<td>4.4 - 6.4</td>
</tr>
</tbody>
</table>

The fuel use charge has been based on purchase prices recently established by the USAEC and a 4 percent per year charge on the value of the fuel allocated to the reactor. The fuel fabrication charge is taken to range from $75 to $125 per pound of fuel; this range of charges is based on General Nuclear Engineering Corporation's estimates for zirconium-clad UO2 fuel rods. The fuel reprocessing costs are based on a value of $10 per pound of fuel, or $20,000 per ton, which represents a small allowance above the AEC charges of $15,300 per ton, to allow for cleanup, loss, and shipping charges. U-6 The charges for net fuel burnup assume credit for the unburned fuel and a $12 per gram credit for plutonium obtained. For $30 per gram credit for plutonium, the fuel costs given in the table would be reduced by about 2.8 mills/kwh.

Figure 7 is a plot of the most recent quoted values of capital costs of nuclear power plants (utilizing plain water as the heat-transfer medium and moderator) under construction or design in the USA. In addition, capital cost values for the British Calder Hall type (gas-cooled) reactors are given. Figure 7 is used to estimate the capital investment, by USA standards, of the Soviet PWR power station. For 210 electrical megawatts per reactor unit, the cost obtained from Figure 7 is about 375 dollars per electrical kilowatt generated. Hence, for the 420 eMW-power station (two reactor units per station), the capital investment required is of the order of 160 million dollars. Hence, for a 14 percent per year charge on
capital investment (USA practice), the cost of power generated by the Soviet PWR station, if located in the USA, is estimated to be as follows:

**TABLE IX**

**TOTAL POWER COSTS**

Soviet PWR, Station No. 1

<table>
<thead>
<tr>
<th></th>
<th>3,500 MWD/ton</th>
<th>10,000 MWD/ton</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plant Investment</td>
<td>7.0</td>
<td>7.0</td>
</tr>
<tr>
<td>Operating and Maintenance</td>
<td>3.0</td>
<td>3.0</td>
</tr>
<tr>
<td>Total Fuel Costs</td>
<td>10.0 – 15.2</td>
<td>4.4 – 6.4</td>
</tr>
<tr>
<td>Total Power Costs</td>
<td>20.0 – 25.2</td>
<td>14.4 – 16.4</td>
</tr>
</tbody>
</table>

A similar evaluation of the graphite-moderated reactor (AP8-1 type) has been made assuming that its characteristics are those in Table II. The remaining unpublished characteristics are assumed to be similar to the published data on their AP8-1 power station. A further assumption is made that fuel element fabrication, being of simpler construction than for their PWR reactor, would cost between $40 and $80 per pound of fuel. These assumptions correspond to a reactor with somewhat more advanced technology than the Soviets will most likely be able to incorporate into their first such graphite-moderated, water-cooled power station. The various items, in a USA accounting system, determining the cost are listed separately in Table X.
TABLE XI
TOTAL POWER COSTS
Soviet APS, Station No. 2

<table>
<thead>
<tr>
<th></th>
<th>2,500 MWD/ton</th>
<th>10,000 MWD/ton</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plant Investment</td>
<td>9.8</td>
<td>9.8</td>
</tr>
<tr>
<td>Operating &amp; Maintenance</td>
<td>3.0</td>
<td>3.0</td>
</tr>
<tr>
<td>Fuel Costs</td>
<td>8.1 - 13.5</td>
<td>4.0 - 6.3</td>
</tr>
<tr>
<td><strong>Total Power Costs</strong></td>
<td><strong>20.9 - 26.3</strong></td>
<td><strong>16.8 - 19.1</strong></td>
</tr>
</tbody>
</table>

In actuality, the first large APS type, Station No. 3, will quite likely require a larger enrichment (up to 3%), and operate with a conversion ratio of about 0.3. Under these conditions, the fuel costs would be:

TABLE XII
FUEL COSTS
Soviet APS, Station No. 2

<table>
<thead>
<tr>
<th></th>
<th>Mills/kwh</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Use Charge</td>
<td>1.2</td>
</tr>
<tr>
<td>Fuel Fabrication and Interest</td>
<td>5.3 - 10.7</td>
</tr>
<tr>
<td>Fuel Reprocessing</td>
<td>1.0</td>
</tr>
<tr>
<td>Fuel Burnup</td>
<td>2.2</td>
</tr>
<tr>
<td><strong>Total Fuel Costs</strong></td>
<td><strong>9.7 - 15.1</strong></td>
</tr>
</tbody>
</table>

This fuel cost, coupled with the lower figure for capital costs of the first such graphite power station (assumed to be the same as for a PWR station, i.e. 7.0 mills/kwh), would yield a net power cost of between 19.7 and 25.1 mills/kwh.
VIII. General Status of Soviet Technology in the Reactor Field

A. Reactor Physics

The Soviet physics effort on nuclear reactors roughly parallels that of the United States. Although there are some differences, particularly in the theoretical approach, they appear to be following a similar program, one that encompasses basic research and applied physics.

Their basic research is concerned with a theoretical and experimental analysis of neutron interactions with matter, and most of their published literature is on that subject.

1. Experimental Program and Measurements

The Soviet experimental facilities appear to be quite good. They have numerous neutron sources which either originate from research reactors or are initiated by particle accelerators or by radioactive sources. Since 1947, the Soviets have constructed and operated a number of low power research reactors. S-1-2 Table XIV summarizes the significant Soviet research reactors. They are fully equipped to measure the necessary parameters for reactor design. Their instrumentation and experimental techniques are comparable to those used at our national laboratories. Most of the Soviet data that have been published involve the properties of fissionable material or assemblies of fissionable material. The results of their cross section measurements, number of neutrons per fission and the ratio of radiative capture to fission capture, are in rather good agreement with our best published data. Whatever disagreement exists is of the same type encountered between laboratories in this country that are measuring the same quantity. Other experiments measuring diffusion lengths and inelastic scattering cross sections also agree with our work to within experimental error.

A considerable amount of experimental work has been done on the physics of heterogeneous lattices. These experiments have been performed primarily on graphite-moderated systems fueled with natural uranium that have been used in both exponential and critical assemblies. The natural uranium-graphite systems perhaps are indicative of a low priority of enriched fuel for experimental purposes. Also, although an exponential facility for light water-uranium lattice is mentioned, S-1-1 no light water-uranium critical assemblies have been reported being used (in the available literature) possibly because of shortage of enriched fuel for such experimental assemblies.

2. Soviet Reactor Theory

The Soviet reactor theorists appear to dominate their reactor physics program, and the experimental work, while considered of great importance, is more in the nature of a supporting operation. The very high percentage of theoretical physics papers presented at the Geneva Conference is indicative of the high esteem the Soviets have for their theorists. Another reason for the dominating role of theory is that the top scientists that
TABLE X
FUEL COSTS
Soviet APS, Station No. 2

<table>
<thead>
<tr>
<th></th>
<th>2,500 MWD/ton</th>
<th>10,000 MWD/ton</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Use Charge</td>
<td>0.5</td>
<td>0.5</td>
</tr>
<tr>
<td>Fuel Fabrication &amp; Interest</td>
<td>5.3 - 10.7</td>
<td>2.4 - 4.7</td>
</tr>
<tr>
<td>Fuel Reprocessing</td>
<td>1.0</td>
<td>0.3</td>
</tr>
<tr>
<td>Net Fuel Burnup</td>
<td>1.3</td>
<td>0.8</td>
</tr>
<tr>
<td>Total Fuel Costs</td>
<td>8.1 - 13.5</td>
<td>4.0 - 6.3</td>
</tr>
</tbody>
</table>

Assuming that the capital cost of an advanced type of large graphite-moderated reactor is the same as for the Calder Hall type, then (from Figure 7) the capital costs of the Soviet power station would be $518 per kw or $104 million total capital investment. At 14%, this contributes 9.8 mills/kwh to the power costs.

The actual capital costs of the first Soviet power station No. 2 would most likely lie somewhere between that for the Calder Hall type and the PWR type. The graphite-moderated reactor (Station No. 2) does not require a pressure vessel as for the PWR type, but would require a fairly large investment for graphite. Furthermore, future developments may be in the direction of permanent pressure tubes of zirconium or beryllium, which, being part of the reactor itself, would contribute to a higher capital investment. These developments would, however, be associated with lower enrichments and would operate with a large conversion ratio. The published characteristics of the APS type power station (Station No. 3) apparently assume that these developments are forthcoming. Hence, the total cost of power generated for an advanced design of the power station is estimated as follows:
The present cost of power in Russia varies from 4 to 12 kopeks/kwh, S-1-9 depending largely upon the cost of transportation of fossil fuel. Near the coal fields, the reported cost of electricity are 4-7 kopeks/kwh, while in Moscow, the cost from coal-fired plants range from 8-10 kopeks/kwh. Since the majority of their energy requirements (70%) are in the eastern part of the country far removed from the Siberian coal mines, nuclear power, with its attendant low fuel shipping costs, offers an attractively low price relative to existing electrical costs in Russia. According to the Belgrade report, S-1-9 the first nuclear power stations being built in Russia will be higher in cost than later stations. Table XIII shows the comparison between present costs and estimated electrical costs from nuclear plants in kopeks/kwh and mills/kwh, based upon an exchange rate of 1 kopek being equal to 2.5 mills. The USSR estimate is based upon the assumption that nuclear power cost is about 1-1/2 times the cost of conventional coal-fired stations.

TABLE XIII

SOVIET POWER COSTS

<table>
<thead>
<tr>
<th></th>
<th>Cost</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>kopeks/kwh</td>
</tr>
<tr>
<td>Coal Plants in Siberia</td>
<td>4-7</td>
</tr>
<tr>
<td>Coal Plants in Moscow</td>
<td>6-10</td>
</tr>
<tr>
<td>Nuclear Power (American Standards)</td>
<td>8-10.5</td>
</tr>
<tr>
<td>Nuclear Power (USSR Estimate)</td>
<td>12-15</td>
</tr>
</tbody>
</table>

The capital costs, as seen from Figure 7, decrease as the plant power level increases. This is due to the fact that the component prices do not increase at the same rate as their size, and that the labor costs also do not increase as rapidly as the installed component sizes. A generally used rule for scaling the cost of plants and machinery as the size changes is that the unit cost increases as the 0.6 power of the capacity. This obviously is only a very rough approximation, especially for nuclear plants, which involve many components. The dashed curve of Figure 7 is a plot of this scaling rule. Obviously it does not hold over a large range, but it is probably reasonably good for extrapolation above the present reactor sizes. The cost estimates for the two British plants follow the rule almost exactly.

The Soviet plants will take advantage of the lower specific price of large plants; this reduces the difficulty of building them so that the cost of the electric power produced at them does not exceed the cost of power at heat plants." S-1-9
guide the programs, such as Blokhintsev and Kurchatov, are apparently theoretical physicists. It appears that the major portion of a reactor design is based on theory with experimental measurements checking specific parts of the theory. On the other hand, the USA position appears to place much more reliance on experiments, particularly the critical assembly type. In fact, most of the USA naval reactors that are in operation were designed primarily by experimental measurements. The Soviets appear to have more confidence in their calculations, although they recognize that there could be considerable errors both in the input data and the theory so that detailed experimental measurements are carried out to verify the calculations.

The basic reactor theory used is quite similar to ours. Our textbook, "Elements of Reactor Theory" by Glasstone and Edlund, U-37 is used quite often as a reference in Soviet papers, indicating it may also be a standard text for them. The primary difference between USSR and USA theoretical methods is one of basic philosophy. Before the USA possessed large computing machines, we tended to make approximations that simplify complex problems, while the Soviet physicists seem to rely on analytical methods, even in the face of extremely lengthy computation. For example, the very difficult problem of computing heterogeneous systems is solved by the Soviets analytically, by attacking the problem in a formal rigorous fashion using numerical methods to achieve the final results for comparison with experiment. S-3-22 These calculations are presumably performed primarily by hand calculation and, as such, involve a tremendous amount of computational labor. There is some mention of an electronic computer being used for lattice calculations involving uranium burnup. S-1-1 but it appears that such computational methods are not yet widely used in reactor physics. However, much of the Soviet information dates back to the Geneva Conference (1955) and no doubt represents work done at a still earlier date; the use of digital computers in the United States has increased tremendously in the last two or three years.

There are some minor differences in reactor theory between the USSR and the United States. For example, the Soviet and USA predictions of the geometric dependence of resonance escape appear quite different. Nevertheless, it has been pointed out by Wigner that the experimental results prove to be very similar to the predictions of both theories over the geometric range of lattices that are of practical interest.

3. Applied Physics

For practical engineering design, the Soviets are willing to sacrifice the exact calculational method and use approximations to facilitate the design. They recognize that reactors are complicated devices whose exact features are difficult to incorporate precisely in any scheme of theoretical calculations. S-2-3 In order to minimize the amount of computation, they use diffusion theory and estimate expected errors both in the mathematical simplification and in the physical constants employed. For
<table>
<thead>
<tr>
<th>Reactor Type (Location) and Power</th>
<th>Fuel</th>
<th>Moderator</th>
<th>Coolant</th>
<th>Affiliation</th>
<th>Date of Criticality</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fersov Graphite (Moscow) &lt;10 kW</td>
<td>Nat. Uranium</td>
<td>Graphite</td>
<td>None</td>
<td>(LIPAN) Lab. of Measuring Instr.'s.</td>
<td>1947</td>
<td>First reactor in USSR.</td>
</tr>
<tr>
<td>Heavy Water No. 1 (Moscow) 500 kW</td>
<td>Nat. Uranium</td>
<td>Heavy Water</td>
<td>Heavy Water</td>
<td>Thermo-technical Laboratory</td>
<td>April 1949</td>
<td>Believed to have been modified in 1954 for gas cooling experiments.</td>
</tr>
<tr>
<td>Swimming Pool No. 1 (Moscow) 300 kW</td>
<td>10% enriched Uranium</td>
<td>Light Water</td>
<td>Water</td>
<td>LIPAN</td>
<td>~ 1953</td>
<td>Testing of shielding materials and configurations.</td>
</tr>
<tr>
<td>First Power 30,000 kW (Obninskoye) 5 eAM</td>
<td>5% enriched Uranium</td>
<td>550 kg</td>
<td>Graphite</td>
<td>Atomic Research Institute</td>
<td>June 27, 1954</td>
<td>Experimental power plant.</td>
</tr>
<tr>
<td>Physical Beryllium (Obninskoye) &lt;10 kW</td>
<td>10% enriched</td>
<td>Beryllium</td>
<td>None</td>
<td>Atomic Research Institute</td>
<td>August 1954</td>
<td>Designed 1953. Contained a Po-Be neutron source.</td>
</tr>
<tr>
<td>Swimming Pool No. 2 (Moscow) 2,000 kW</td>
<td>10% enriched Uranium</td>
<td>Light Water</td>
<td>Water</td>
<td>Moscow State University</td>
<td>~ 1955</td>
<td>13 Test loops. Neutron (Basic) research.</td>
</tr>
<tr>
<td>Plutonium Fast (Obninskoye) 200 kW</td>
<td>Plutonium 12 kg</td>
<td>None</td>
<td>Mercury</td>
<td>Atomic Research Institute</td>
<td>Feb. 1956</td>
<td>First Metal Cooled Reactor.</td>
</tr>
<tr>
<td>Heavy Water No. 2 (Moscow) 2,000 kW</td>
<td>2% enriched Uranium</td>
<td>Heavy Water</td>
<td>Heavy Water</td>
<td>Thermo-technical Laboratory</td>
<td>1956-1957</td>
<td>Modification of Heavy Water No. 1 to obtain a higher power &amp; flux.</td>
</tr>
<tr>
<td>Sodium cooled fast (Obninskoye) 5,000 kW</td>
<td>Plutonium 50 kg</td>
<td>None</td>
<td>Sodium</td>
<td>Atomic Research Institute</td>
<td>1957</td>
<td>First sodium cooled reactor.</td>
</tr>
<tr>
<td>Research (Dubna) 1 kW</td>
<td>---</td>
<td>---</td>
<td>---</td>
<td>Dubna</td>
<td>---</td>
<td>You experimental lab.</td>
</tr>
<tr>
<td>Research (Georgia) 1 kW</td>
<td>---</td>
<td>---</td>
<td>---</td>
<td>Georgia</td>
<td>---</td>
<td>Possibly homogeneous.</td>
</tr>
</tbody>
</table>
example, in their APS-1 (5 MW electrical) reactor they use a simplified two-group theory in which the input data are determined by empirical methods. S-2-4 However, they checked the calculational accuracy by using a multigroup method. S-2-5

As an indication of the uncertainty in the Soviet calculations, the results of a measurement on a critical assembly, composed of graphite-water moderator and 5% enriched fuel, showed criticality was achieved with 50-54 fuel elements, depending on the condition of the core. The calculations implied a loading of 42-56 fuel elements, the wide range accounting for the limits of accuracy. S-2-4 Such numbers are quite consistent with the large uncertainties experienced in USA reactor calculations.

4. Reactor Physics Problems

Since there is so much similarity between the USSR and USA reactor programs, it is not too surprising that the countries are working on similar reactor physics problems. For example, at present the Soviets are quite interested in light water-uranium reactors. In this connection, they are striving to acquire more accurate physical constants, study more details of the neutron slowing-down process, and gain further knowledge concerning the inclusion of plutonium as a fuel in such reactors. S-1-1 In connection with the use of plutonium, they are doing a considerable amount of theoretical and experimental work on epithermal captures in plutonium, as well as the determination of the neutron energy spectrum in the thermal energy range. Such problems are identical to those with which our reactor physicists are concerned. Undoubtedly, part of the similarity stems from information that evolved from our program. However, it should be remembered that the reactor field is such that competent people working on similar programs would be expected to encounter similar obstacles. In general, the over-all character of the Soviet physics program appears to resemble the early American program in that the major effort places great emphasis upon the theoretical approach.

In its rather theoretical approach, the Soviet reactor physics program is reminiscent of the USA program of a few years ago. It is characteristic that a nuclear energy program in any country is given the benefit of outstanding theorists in its early years, and that in later years workers who lean more toward the engineering approach take over the work of reactor design, building on the foundation established by the theorists. The difference between the present Soviet situation and the situation which existed in the earlier days of the USA program is that a great deal more experimental and technological information is now available. This may emphasize the contribution which the theorists are able to make to the Soviet program.
B. Fuel Technology

Probably the most important aspect of economic nuclear power is the development of suitable fuel elements for power reactors. Research having to do with fuels appears to be an extremely important phase of the USSR program.

1. Uranium

In 1950-52, S-1-10 the Soviets were studying various means of electrolytically cladding uranium with iron, chromium, copper, nickel, and silver. These experiments were not successful and further tests were conducted with uranium enclosed in stainless steel, using liquid sodium as a bond.

The irradiation damage and dimensional instability of uranium were first noted by the Soviets in experiments conducted in the first Russian reactor, S-4-1. In Blokhintsev's paper on the atomic power station, S-3-5 he discussed briefly the metallurgy of uranium metal, including the transition temperatures and the dimensional instability of uranium when thermal cycled through these phase transitions. His discussion of the effect of irradiation on the physical properties of uranium indicated considerable metallurgical studies had been made on this material, including the studies of the effect of irradiation on the creep properties of uranium. He also indicated that the fuel element was a special uranium alloy which was plastic during irradiation and would deform easily so that good thermal conductivity would be maintained without rupturing the fuel can when volumetric changes took place. At the time of this conference, it was stated that no fuel element failures had occurred since operation of the reactor had started on June 23, 1954 (15 x 10^6 kwh of electrical generation).

Some of the studies made on uranium up to 1955 were presented in a paper at the Geneva Conference on the irradiation behavior of fissionable materials. S-3-28 This paper covered the various physical property changes of uranium under irradiation as well as some of the irradiation effects of uranium-molybdenum alloys. The data presented at the Geneva Conference revealed good knowledge of the technology of uranium, but did not indicate that an extensive program had been undertaken to develop uranium alloys. Since the Geneva Conference, the USSR has very probably improved its technology of uranium alloys.

2. Thorium

Although thorium-232, which occurs naturally, is not a fissionable material, it can be converted in a reactor to uranium-233 which is fissionable and therefore is useful as a reactor fuel. Very little has appeared in the Soviet literature about the technology of thorium. A paper presented at the Geneva Conference on the metallurgy of thorium indicated that studies were being made. S-3-29
Kurchatov at Harwell S-1-1 mentioned that thorium behaved much better in reactors than uranium, even when large amounts of U-233 had grown into the Th-232.

As part of the experimental reactor power station, a thermal homogeneous D2O-moderated reactor is to be constructed. S-1-9 It is expected that a Th232-U233 cycle may be used eventually.

Other papers have been presented on the investigation of fused salt systems based on thorium fluoride. S-2-24, S-2-22 Although these studies would be for processing systems, the results can be applied to fused salt reactor systems.

From the above information, it is evident that the Soviets are expecting to use thorium eventually and probably have a study program on this material. A large amount of information on thorium is available to them in the unclassified USA literature.

The technology of thorium with produced U-233 will be more important than that of pure thorium for a reactor program. The problems of chemically separating and refabricating into fuel elements the highly radioactive U-233 is a major problem common to both the USA and the USSR. There is nothing in the Soviet literature to indicate that a satisfactory solution to this problem has been found.

3. Urania (UO2)

Probably the first Soviet mention in unclassified literature of UO2 as a fuel for reactors was the paper presented by Alkhanov at the Geneva Conference. S-3-7 This concerned the use of a UO2 suspension in water for a homogeneous water boiler reactor proposal. Since then, Kurchatov at Harwell stated that pressed and sintered UO2 elements were very stable under irradiation and that very few fission products escape from unclad UO2. S-1-1 This would indicate that the UO2 elements were of fairly high density, 90-95% of theoretical, and that they were irradiated to relatively low burnups, probably of the order of 2000-3000 MWD/ton. Combining this information with the fact that the Soviet power program includes two 420 eMW pressurized-water type reactor stations using zirconium-alloy jacketed UO2 fuel, S-1-9 it appears that the USSR has accelerated its investigations of the technology of UO2 since the presentation by the USA at the Geneva Conference of the use of UO2. U-16, U-17

The Soviets at Belgrade S-1-9 have also indicated that burnup of the fuel elements for their PWR can probably approach 10,000 MWD/ton. This information has been known in the USA for some time and the data on burnups of UO2 of this order are declassified.

The presently available unclassified information from the USSR indicates a slight lag in their technology of UO2; if such a lag existed, a real effort has probably been made to eliminate it because of the proposed use of UO2 in the PWR reactor.
tions, the form of which are derivable in many cases from simplified theories such as Prandtl's and Von Karman's mixing length theories. The Soviet work published so far uses the same experimental and semi-empirical correlation approach.

In regard to their experimental approach in the liquid-metal work, the equipment, methods used, and care taken in obtaining and interpreting the data, are of high quality, and certainly comparable to the quality exhibited in research laboratories in the USA and England. It is interesting to note that the only experimental liquid-metal heat transfer work available in the USA for many years was that performed in the USSR as early as 1936-1940. S-5-8 The first available liquid-metal heat transfer data of any reliability were not obtained in the USA or England until as late as about 1947. U-31, U-32 The early Soviet data agree reasonably well with the mass of data now available on liquid-metal heat transfer.

At the Geneva Conference the Russian presentation on liquid-metal heat transfer was rather definitely superior to that of the United States. The USA presentation was not typical of the best heat transfer technology available in the United States, but it was perhaps typical of that which goes into reactor design in this country. It may well be that in the applications of heat transfer to reactor design, the Soviets are more highly skilled. From the published information relating to their water-cooled reactors, it would appear that they are bolder in approaching the upper limits of attainable heat transfer performance.

D. Liquid Metals

Extensive studies have been conducted in the USSR on liquid metals for use in reactors. S-1-10 The main concentration of effort was on sodium, sodium-potassium eutectic alloy (NaK), and lead-bismuth alloy. Experiments were conducted on the diffusion of sodium, potassium, and bismuth alloys into uranium metal at temperatures as high as 600 to 800 °C. S-1-7 The Russian papers presented at the Geneva Conference in 1955 on liquid metals indicated that at that time the basic scientific and engineering knowledge necessary for the building of metal-cooled reactors was well advanced. S-3-23, S-3-24, S-3-26 These papers showed the USSR was conducting: (a) extensive studies on the production of high purity metallic bismuth; (b) investigations of the high temperature heat transfer properties of mercury, tin, lead, bismuth, sodium, sodium-potassium eutectic alloys, and bismuth-lead alloys; and (c) investigation on the properties of iron alloys in lead, bismuth, and lead-bismuth eutectic alloys in the temperature range of 500 to 600 °C.

In a more recent paper, the Soviets have presented further information on various properties of the liquid metals. S-2-17 The investigations included: the heat transfer properties of liquid sodium in turbulent flow through copper and nickel tubing; determination of the thermal resistance at the interfaces of liquid sodium and solid copper, nickel, and stainless steel; and measurement of the viscosity, thermal diffusivity, and density of sodium, potassium, lithium, and sodium-potassium eutectic up to temperatures of 800 °C.
4. Plutonium

Very little can be said of the USSR technology on plutonium. This information is probably classified as military information. Since it is quite obvious that the USSR has atomic bombs, a significant body of plutonium experience undoubtedly exists. This is further confirmed by the fast reactor program, S-2-32, S-2-33

C. Heat Transfer

One of the important limitations to the permissible power output of a given size reactor is the heat removal capability of the heat transfer medium used. For this reason, there has been a significant increase in effort in the USA on experimental and analytical heat transfer research work. Recently published papers indicate a comparable increase in the USSR on reactor heat transfer work. This, of course, is to be expected as a natural development of the Soviet plans to study and develop economically competitive nuclear power stations.

The Soviet work in heat transfer appears to parallel the work in the USA. In connection with the development of the pressurized water reactor, work is evidently under way on the crucial problem of determining, and avoiding, if possible, the so-called burnout limit to the maximum heat flux which can be obtained for the transfer of heat from solid surfaces to flowing water. A large amount of work on this problem is in progress in the USA.

Paper P/639 3-3-28 presented at the Geneva Conference in August 1959, and a recent paper S-2-17 indicate the continuing interest of the Soviets in the field of liquid-metal heat transfer. This work no doubt represents a supporting effort in the design of the 50 eMW graphite-moderated, sodium-cooled reactor, and the 50 eMW sodium-cooled fast reactor. In both these reactors, a sodium-potassium eutectic alloy is used in a secondary circuit as a means of transferring the heat from the reactor-radioactive sodium to the steam finally used to run the turbines.

In view of the Soviet plans to construct a boiling water reactor, work is probably being done on the flow and heat transfer characteristics of steam-water mixtures in closed channels.

Judging from the published Soviet papers on liquid-metal heat transfer, their approach to heat transfer research problems appears to be identical with the USA approach. In heat transfer and fluid flow problems, the approach, on the basis of pure fundamentals, has not been particularly fruitful in the USA or England. This is because of the incompleteness in the understanding of the mechanisms of turbulence and vorticity associated with the development of boundary layers along heat transfer surfaces. In the USA and England, some encouraging work has been in progress on turbulence theory, but it is not known what work the Soviets are doing in this respect. Because of the long-range nature of such fundamental work, the approach in practical heat transfer problems has been primarily experimental. The experimental results are usually correlated by empirical equa-
Not much has appeared in the unclassified USSR literature on in-pile test loops using liquid metals. It would be expected that such tests would be made in connection with the development of the two proposed metal-cooled reactors, S-1-9. These are the 50 eMW graphite-moderated and the 50 eMW fast breeder reactors, both of which are sodium-cooled. The USA has operated a number of sodium coolant loops, some quite large, as well as several reactors cooled by liquid sodium (EHF-1, Experimental Breeder Reactor; SRE, Sodium Reactor Experiment; SIR, Submarine Intermediate Reactor).

The Soviet investigation of the bismuth-lead eutectic alloys may indicate an interest in liquid-metal fueled reactors. Extensive work has been done in the USA in the past few years on metals for use in liquid-metal fueled reactors, and a number of large experimental liquid-metal loops using Pb-Bi have been operated.

Although the laboratory work on liquid metals in the Soviet Union appears to be quite extensive, there is no indication of large-scale use of liquid metals in circulating systems. Even if extensive loop tests and mock-up tests for liquid-metal cooled reactors have been made, the experience gained will hardly be comparable to that which has been obtained in the United States in the actual operation of metal-cooled reactors. It is therefore probable that although the basic knowledge of liquid metals and liquid metal handling may be extensive in the USSR, the practical technology of large-scale systems, and particularly of irradiated systems, lags considerably behind that of the United States.

E. Heavy Water (D₂O)

Information from the USSR indicates they were interested in the use of D₂O as far back as 1948, S-1-7 and probably well before that time. At the Geneva Conference in 1955, D. I. Blokhintsev S-3-6 indicated that one of the disadvantages of power reactors using heavy water was the high cost of the D₂O. At that time, while studies of this moderator material were going on in the USSR, the cost was probably too high and production too small to warrant its use in competitive nuclear power reactors, although D₂O has been used in at least one research reactor. Heavy water power reactors were proposed S-1-4, S-3-7 by the USSR at the Geneva Conference in 1955. Also, Kurchatov S-1-1 mentioned that one of the experimental reactors to be built as part of the power program will be a homogeneous, D₂O-moderated reactor.

It has been recently reported that the USSR has two plants producing heavy water, U-11. One produces 60 tons per year, using a hydrogen sulfide exchange system in which the water is enriched to about 5%. This is followed by an electrolysis process producing 99.8% D₂O. The second plant uses an ammonia-hydrogen exchange process with a capacity of 30-40 tons per year. This total of approximately 100 tons per year would indicate that no large nuclear power plant using D₂O as the moderator is planned in the near future.
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The USA production of D₂O has been large enough for this country to sell its plant in Canada U-19 and to discontinue production at its Indiana plant. U-19 This leaves only the Savannah River Plant, which has sufficient capacity to furnish all the D₂O necessary at a price of $28 per pound.

Although the Soviet production of D₂O is not nearly as great as that in the USA, there is nothing to indicate that their knowledge of D₂O physical and chemical characteristics is not on a par with ours. However, the small production of D₂O would necessarily have limited the practical experience gained in the technology of D₂O.

F. Zirconium

The Soviet knowledge of zirconium technology prior to 1955 had apparently not reached the same level of development as that of the USA. The only zirconium paper presented by a Russian at the Geneva Conference in 1955 was one by Poluektov of the Ukrainian S. S. R. S-3-3 on the chemistry of zirconium and hafnium. However, in Kurchatov's address to the Twentieth Congress of the Communist Party in March of 1956 S-5-6 he mentioned the development of new metals, probably Zr, for use in the proposed reactors of their nuclear power program. Even prior to this date, there was indication that the USSR was producing hafnium-free zirconium and that independent research was being done on zirconium. S-1-11

Since the Geneva Conference in 1955, the USSR will have had access to much of the declassified material on zirconium and its alloys, such as the book by B. Lustman and F. Kerse, Jr., on "The Metallurgy of Zirconium" (1955), and the USABC Reactor Handbook Volume on Materials (1955).

Information has recently been released by the USSR that the fuel element clad and fuel element boxes for two of their large pressurized water reactors would be made of a zirconium alloy. S-1-5, S-1-8, S-1-9. Such expected use of zirconium would mean that the technology of processing zirconium sponge, the metallurgy of various zirconium alloys, and the fabrication techniques of zirconium alloys are being studied and developed. As part of such a program, the irradiation behavior and corrosion resistance of zirconium and zirconium alloys would also be studied.

Further indication that the USSR has been working extensively on zirconium technology is obtained from the number of papers being released in the Soviet journals. In this year, 1957, five (5) papers have appeared in the literature concerning:

1. the formation of zirconium sponge in the Kroll Process; S-2-32c
2. the study of the strength and air oxidation resistance properties of zirconium-niobium alloys at elevated temperatures up to 750 C; S-2-32a
3. the measurement of the Young's Modulus of zirconium-niobium alloys up to temperatures of 950 C; S-2-32b

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4. an investigation of the iodide method of refining zirconium; S-2-33
5. studies of the metallography, X-ray diffraction, thermal analysis, electrical resistance, hardness, and other physical properties of the zirconium-tantalum alloy system. S-2-31

There is some evidence for believing that the Soviets may have developed a zirconium-aluminum alloy which is satisfactory for fast reactors. If so, this would indicate that a high-temperature alloy, corrosion resistant to Na or NaK, has been developed. Although the majority of the work in the USA was done on zirconium-tin alloys (Zircaloy) for use in water reactors, some studies were also conducted on zirconium-aluminum alloys. The zirconium-aluminum alloys have shown good corrosion resistance and strength in sodium at temperatures as high as 1000 F. U-20 Perhaps for this reason, the Soviets are planning to use this alloy as the canning materials in their sodium-cooled fast reactors and in their sodium-graphite type reactors.

Because of the large use of zirconium alloys contemplated in the power plants proposed at Belgrade, S-1-9 it is quite evident that the technology of zirconium is being studied extensively in the USSR. The USA has had a large amount of working experience with the production, fabrication, and behavior of zirconium and its alloys, due mainly to the government's atomic powered submarine and surface vessel program. More recently, the high cost of producing and fabricating zirconium has led some private companies in the USA to abandon zirconium in favor of stainless steel for H2O-moderated reactors. The USSR, on the other hand, because of its government-controlled reactor programs, is probably emphasizing the studies on zirconium and zirconium alloys. There is little doubt that zirconium will continue to be one of the most important metals in the nuclear power field.

G. Graphite

Graphite was used in the first reactors because of its availability, good neutron moderating properties, and low neutron absorption cross section. Ordinary graphite, however, contains contaminants of high neutron absorption cross section and must therefore be processed to produce reactor-grade graphite. The USSR was conducting graphite reactor studies at Omskoye S-1-10 prior to 1947 and successfully operated their first graphite-moderated reactor in 1947 at the Academy of Sciences in Moscow. This experimental graphite-moderated pile was used as the prototype for the 100-200 megawatt plutonium production reactors, S-1-9 in which the fuel channels are believed to have been placed vertically and cooled by pressurized light water.

The application of graphite to their production reactors brought with it all of the problems having to do with the radiation damage to graphite. Understanding of these problems by the USSR was clearly evidenced by their presentation of a paper prior to the Geneva Conference on the changes in the properties of graphite when irradiated. S-4-16 This
paper covered very comprehensively the experimental evidence on reactor-grade graphite concerning:

1. The variation of electric resistivity with total neutron absorption.
2. The effect of irradiation temperatures on the resistivity.
3. The effect of thermal annealing of graphite on the resistivity.
4. The changes in the specific volume of graphite as a function of radiation dose.
5. The effect of temperature on these volumetric changes.
6. The effect of thermal annealing on the irradiation growth as a function of temperature.
7. The variation of thermal conductivity with irradiation.
8. The effect of irradiation temperature on the thermal conductivity.
10. The accumulation of stored energy in the graphite during irradiation.
11. The release of stored energy as a function of the thermal annealing temperature and the total radiation dose.
12. The variation in the Young's modulus and hardness as a function of irradiation dose and temperature.
13. Crystal lattice parameter changes with radiation dose.

It is quite clear from the amount of detailed and accurate information presented in the paper S-4-16 that the irradiated graphite technology was well understood.

The use of graphite in the first USSR atomic power station, S-3-6, in their plutonium production reactors, and in their 400 eMW power station now under construction S-1-9 is clearly indicative of the continued progress of the USSR in the technology of graphite.

It appears that both the USA and the USSR have all the knowledge of graphite necessary for the construction of large graphite-moderated reactors. Studies are continuing in the USA, England, and France on improvements in the nuclear and physical properties of graphite. Two improvements which have been reported are: (a) the decrease in the effective absorption cross section of graphite by increasing its purity; and (b) the increase in density by better fabrication processes.

No evidence is available from the USSR that such improvements in their reactor grade graphite have yet been made.

II. Beryllium and Beryllia

Because of the very low thermal-neutron-absorption cross section and high neutron scatter cross section of beryllium (Be) and beryllia (BeO), it has excellent potential use as a moderator and reflector for reactors. In addition, both the metal and the oxide have high thermal conductivities, high melting points, good high-temperature strength, and are light in weight. The main disadvantage of beryllium metal is its low ductility.
making fabrication difficult. In addition, the corrosion behavior of Be in hot water has been erratic, with some specimens more corrosion resistant than zirconium while other specimens disintegrated completely. Its toxicity is also a disadvantage. Beryllia, as is the case of most ceramic material, is very brittle but is corrosion resistant to hot water.

The USSR's interest in beryllium started early in their studies of nuclear reactors. There were reports of spectrographic analysis of beryllium samples as early as 1946. S-1-12 In 1950, the Soviets took over most of the research on beryllium from the German scientists at Omskoye and started a series of studies on beryllium and beryllia. S-1-7 Preliminary calculations were made at that time on a one megawatt beryllia (BeO) reactor using 2.8% enriched uranium with helium cooling. It was estimated that at the end of 1952 approximately 3,300 pounds of Be and BeO were available. Some of the interest in Be and BeO at that time was possibly due to its favorable nuclear properties for use in a plutonium-production reactor facility and the use of Be in conjunction with polonium and sodium as a neutron source. Studies of the same general nature had been made earlier in the USA; the design of a BeO-moderated reactor, the Daniels Pile, was worked on intensively, and considerable research and development was carried out. U-21

The extensive study made on Be and BeO in the USSR prior to 1955 is indicated by the papers presented at the Geneva Conference in August of 1955. S-3-18, S-3-19, S-3-31 These papers presented some of the nuclear physics studies conducted, such as the measurements of thermal neutron capture cross section, neutron age, and neutron diffusion length, as well as some of the technology of manufacturing of pure beryllium and beryllium oxide for use in reactors.

The early production of pressed and sintered BeO by the USSR indicated excellent technology in the field of ceramics which would be required to obtain large, dense, pure specimens of BeO.

In 1954, a beryllium metal-moderated reactor was put into operation S-2-20 to study the properties of such a reactor. Approximately 545 pounds of pure Be metal was used in this reactor. Individual metal blocks measuring 6.35" x 6.35" x 1.5" and having a density of 1.78 g/cc were fabricated.

The density of these Be blocks is low compared to the theoretical density of 1.847 g/cc, indicating that they were probably made by a simple powder metallurgy technique as indicated in reference S-3-31. In reference S-3-31, beryllium specimens prepared by more complex powder metallurgy techniques and densities of 1.84 to 1.85 g/cc which is essentially theoretical density.
rate of radiolysis of water with variations in H₂ and O₂ concentrations, reactor power, contaminants and temperature and pressure.

These experiments employed some techniques borrowed from American investigators and in most cases extended the range of the variables studied. On problems on which data in the literature are lacking, such as the influence of dissolved O₂ on the saturation pressures of H₂, as well as the question of the influence of acidity on the rate of radiolysis, the Soviet experimenters have devised techniques to overcome the difficulties of such measurements which they claim has impeded the work of other investigators.

The mechanism of water radiolysis proposed by an American U-24 was checked with their experimental results with good accuracy except in cases of high oxygen concentrations in water, in which case the theory was felt to be inadequate and a supplementary equation was devised.

From the results of this work, and their development of water-cooled reactors, it appears that Soviet developments in this field are at least equal to or more advanced than those in the USA.

Operation of a number of research reactors (see Table XIV) as well as the large water-cooled plutonium producer reactors, has given the Soviet Union opportunity to gain practical experience in the technology of both heavy and light water at low temperatures.

Quite evidently, the technology of water purification by ion exchange resins is not well developed in the USSR. Prior to the Geneva Conference, water purity was maintained in all of the Soviet water-cooled reactors either by distillation or by purging. Evaporators are used for maintaining water purity in the proposed pressurized water type reactor. The possibility of using ion exchangers is mentioned, but it is stated that evaporators are more reliable. All indications in USA experience are that ion exchange is such to be preferred if suitable and reliable resins are available and if the price for such resins is not exorbitant.

J. Containment and Safeguard

From the literature available, there is little or no containment for reactor installations. Protection against explosion or other violent releases of energy (e.g. metal-water reactions) within the reactor vessel that may yield fission products to the atmosphere are not mentioned in their safety programs. The exception to this is a statement citing that power stations would not be built in cities because of the inexperience with the operations of some atomic power producing plants. S-1-9 The reactors are thought safe, but for the present would be built well outside of populated areas. On the other hand, Kurchatov said S-1-1 over a year earlier that they anticipated safe operation of atomic power stations and were ready to build them near large towns. Another reference from the Geneva Conference indicates from a question by Krzhizhlin S-3-33 that the Soviets were not anticipating containment problems because of the inspection procedures and factors of safety worked into power plant design.
There has been very little information in the past few years on the USSR technology on Be, Be alloys, or BeO. This may be due to the cost of billets of beryllium metal, which in 1955 was approximately $90 to $125 per pound in the USSR, U-21 but, more than likely, it has been abandoned in favor of zirconium because of the difficulty of manufacturing Be into shapes suitable for a fuel-canning material, and the fact that it does not have any great advantage over zirconium for large H2O-moderated reactors which the USSR is constructing.

In the USA and in England, further studies on the behavior of Be in water are being conducted. U-22 The USAEC has recently contracted for the production of a million pounds of beryllium at a price of $37 per pound, to be used probably for continued experimental studies on this material. U-23

Beryllium metal would find great usefulness as a canning material for high-temperature, gas-cooled, natural uranium reactors such as the British Calder Hall type reactors for which its high-temperature and nuclear characteristics are excellent. The Soviets have proposed a gas-cooled, heavy-water power reactor S-1-4 in which the light metal can on the fuel could withstand a temperature of 550°C. Aluminum or magnesium alloys cannot be used at this temperature, and zirconium and its alloys are not corrosion resistant to the high-temperature coolant gases, such as CO2 and air, proposed for use in such a reactor. This would leave either beryllium or stainless steel as possible materials for the can. Of the two, beryllium has the most favorable nuclear properties for use in this reactor; in fact, the reactor could not operate as proposed with natural uranium if steel were used as the can material. Later information on the USSR power plants to be constructed do not show a gas-cooled D2O-moderated reactor, S-1-9 indicating this project may have been postponed pending either the development of a suitable canning material, such as beryllium, or until sufficient D2O production is available.

Because of their favorable nuclear characteristics, Be, Be alloys, and BeO have been considered as major nuclear materials during the early stages of reactor development in most countries. However, in view of some of the unfavorable physical properties such as the low ductility and corrosion resistance, these materials have more recently been considered primarily for more specific purposes, such as reflector materials for compact reactors and fuel jackets for gas-cooled reactors. There is no doubt that further studies similar to those being conducted in the USA are also being made in the USSR, and solution to the physical property disadvantages of Be or the successful application of Be, BeO, or Be alloys may have a major effect on the reactor program of the USSR. Such an application to consider would be a Be canned, natural uranium, high-temperature, gas-cooled, BeO or D2O-moderated reactor.

I. Water

A report of Soviet work done in 1950 S-3-32 on the radiolytic decomposition of water under reactor operating conditions indicates a very extensive investigation of this problem. Their work includes a study of the
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By American standards, the proposed 760 MW pressurized-water reactor would require containment if built in or near a populated area. The core of the reactor is filled with Zr alloy jacketed fuel assemblies. The water and zirconium with the uranium heat source represent a known explosive mixture under the proper conditions. Although it is unlikely that such an explosion would take place even in abnormal operation of a reactor, high pressure peaks of short duration comparable to TNT detonation are possible.

In addition to the savings in cost of the expensive containment building, another advantage in having no reactor containment comes from the grouping of reactors in a large central station. Several reactors may share a single building and equipment as do typical coal plants. This is very economical, whereas the necessity of containment indicates that each reactor be in a separate container, where sharing of expensive heavy equipment is impracticable.

To date, there has been no indication in the available Soviet literature to show that they may have conducted any experiments in reactor safety. Several experiments to determine the safety aspects of reactors have been undertaken in the USA. Although experiments of this type, such as Borax, and the homogeneous excursion experiment, are very important in reactor safety, the Soviets appear to rely heavily upon published American information, and on theoretical prediction. In general, except with regard to containment structures, their reactor safeguard procedures and design principles appear to be comparable to those prevalent in reactor design in the USA.

K. Pressure Vessels

The foremost item for discussion of pressure vessels is the Soviet 760 megawatt pressurized-water reactor vessel. This vessel is large in comparison to existing USA vessels and to any contemplated USA designs of 1500 psi rating. This vessel S-1-9 is 11' 10" inside diameter and 39.4' high, operating at 1470 psi - 52°F. The wall thickness is 3.93" for straight cylindrical stress and 7.09" thick at the nozzle junctions. The material used is said to be high strength heat resistant steel with a yield point of 71,200 psi at 617°F. The lid with control rod penetrations is flat, and wedge gasketed in a self-sealing manner, using 5-1/8" bolts in a bolt-sealing ring.

At Belgrade, S-1-9 in spring 1957, and in the U.N. reply, S-1-8 the Russians reported their work on this vessel, and mention was made of tensiometric models. These models will be used for studying stresses experimentally as well as theoretically. Internal structures were being investigated experimentally with models also. The 760 MW vessel design is probably well under way with a maximum effort to achieve a major goal in building a successful vessel of this size. An effort of this magnitude probably does not exist in USA civilian activities. The largest USA vessel under design contract is 12 ft. outside diameter and 40 ft. tall. It is to operate at 1000 psi, 500 psi less than the Soviet vessel; yet the wall thickness is 5-1/2 inches, 1-1/2 inches more than the Soviet specifications.
The material used by the Russians is said to have a yield strength of 71,200 psi at 617°F; the design stress incorporates a factor of two to yield strength, setting it at 35,600 psi. A brief comparison of this material to existing USA materials commercially available with ASME Boiler Code ratings are SA 212 B, 26,000 psi at 650°F, and SA 302 B, 45,000 psi at 750°F (yield strengths). It is clear that the Soviet material exceeds the figures given, but the bases for comparison are unknown; that is, maximum or minimum yield strengths, ultimate strength, etc. Materials having similar strength and properties are chromium-molybdenum steels, properly heat-treated. Chromium-molybdenum steels can be had with 105,000 psi minimum ultimate strength at 650°F and, in accordance with the ASME Code, have design strength of 26,500 psi. Such a steel is among those being tested in the USA, with a good possibility that it will soon become a commercial product. It is presumed that the Soviet steel is of the chromium-molybdenum variety.

The use of a high strength steel is not a necessity, but effects a savings in vessel material, due to thinner vessel wall thickness, as well as reducing the thermal stress problem from neutron and gamma radiation. An example for comparison in the USA would be the Dresden vessel, which is equal in over-all dimensions. The wall thickness of the Dresden vessel, if extrapolated to a pressure of 1500 psi for comparison would be about 6.5 inches (including a 10% reduction in design stress for thermal stress) as compared to the 3.98 inches of the Soviet vessel. Such a vessel (Dresden) is feasible to build with USA commercial steel SA 302 B.

Transportation of large vessels can be a problem. The weight and dimensions, if limited to standard railways almost anywhere in the USA, would be about 11' in outside diameter x 45' long, weighing 300 tons (two flat cars). Larger vessels must be shipped by barge or, if not much larger, can be shipped by rail, on special routing.

The lid of the Soviet vessel displays a unique and rather difficult approach to such a large vessel. It is a flat plate, gasket-sealed, with control rod drive penetrations. For the thick plate that would be necessary for such a lid, it would be quite difficult to acquire a uniform hardness and strength, while the penetrations would tend to misalign, due to any bending of the lid. The gasket is not shown as welded and, as a non-welded gasket, it would be difficult to seal. In the USA, gaskets to 8 ft. diameter have been attempted unsuccessfully. A 6-1/2 ft. gasket has been successful, but an 11-1/2 ft. opening seems doubtful to seal without welding. There are no apparent refueling ports, so any fuel handling would mean complete disassembly of the head; thus the gaskets must be practical and positive.

The remaining portions of the pressure vessel show no apparent uniqueness to similar proposed vessels in this country.

Summing up the Soviet vessel, it shows that a great effort is being put forth to engineer a good and very large pressure vessel. Stress analysis and metallurgical work appear to be well advanced. This vessel would
probably represent the first of a production-line model for pressurized water reactors, and, with modifications, boiling water reactor vessels can also be made using the design of this vessel.

The Soviets have not shown much preference for large pressure vessels of the Calder Hall type. Whether this is based on a question of technical feasibility or cost is not known. Such pressure vessels have not been made in the United States, but a design study leading toward such a reactor is currently under way. There is little doubt that such a vessel can be fabricated in the United States, but its cost may be high.

L. Radiation Effects

Because of the physical and chemical property changes that occur when materials are subjected to irradiation, the study of various radiation effects on materials becomes a very important phase of reactor development.

Such studies require fairly elaborate and expensive facilities for an adequate job. Among these facilities are "hot" (radioactive materials) laboratories and research reactors. The Russian papers presented at the Geneva Conference in 1955 indicated a keen appreciation of the radiation effects on materials. Among the papers presented at that meeting which gave an estimate of their irradiation program up to that time were:

1. P/673 S-3-33

   This dealt with the metal research "hot" laboratory for the study of physical and mechanical properties of irradiated materials as well as for the metallographic preparation and examination of materials. The hot laboratory consisted of six major cells: (1) A "hot" machine shop cell capable of handling materials having a radium equivalent activity of up to 50,000 curies. This would be sufficient to handle complete fuel assemblies from a reactor. (2) A transfer and storage cell. (3) A metallographic cell for the preparation of specimens. (4) and (5) Two physical measurement cells, fully equipped. (6) A mechanical testing cell complete with remotely-operated test apparatus.

   It is interesting to note that the design of this hot laboratory showed excellent contamination control and radiation hazards protection. Remotely-operated apparatus was used throughout and master-slave type mechanical manipulators were used for general purpose handling. Viewing was accomplished by periscope mirror arrangements and by shielding type lead glass windows. This facility is comparable to most of the best designed "hot" laboratories in the USA. The number of such laboratories in the USSR is not known.

2. P/672 S-3-34

   This paper dealt with a "hot" laboratory facility similar to the metal research hot laboratory, but especially suited to handle radioactive chemicals. The heavily shielded cells were designed to handle as much as 10,000 curies, radium equivalent, of specimens irradiated in their reactors.
Other cells were designed for millicurie work, while still others were equipped with shielded glove boxes allowing manual handling of the chemicals.

Again, excellent contamination control is evident in the Russian design. Remotely-operated mechanical master-slave manipulators, shielding type lead glass viewing windows, remotely-operated apparatus, stainless steel lined walls, special "hot" drains, remote specimen transfer systems, shielded hoods, mobile shields, and other apparatus were used.

It can be stated that the design of this "hot" chemistry laboratory is on par with those in the USA.

In all irradiation programs, research reactors in which the materials are placed to be irradiated under controlled conditions are required. The USSR has constructed at least ten reactors, S-1-2 shown in Table XIV, which have been used for irradiation experiments. In addition to these research reactors, the first USSR Atomic Power Station S-3-6 as well as other production reactors have been or could be used for irradiating specimens.

At the time of the Geneva Conference, the research and test reactors existing in the USSR represented a fairly impressive test capability. Since that time, the test capacity in other countries with large nuclear programs has been expanded considerably. The United States has added the Engineering Test Reactor, the Oak Ridge Reactor, and a number of small research reactors to its test equipment which was already outstanding. The United Kingdom has added, or is adding, at least three 10 megawatt test reactors, and Canada is building a very large natural uranium research and test reactor. A complete knowledge of test reactors presently available and under construction in the USSR would be quite helpful in assessing the effort which is being put into the atomic energy program. The only new reactors of this type which have been announced are the reactors in Dubna and in Georgia, whose power level has not been stated, and the sodium-cooled fast reactor, which is primarily a reactor experiment rather than a test reactor.

Some of the irradiation tests reported at the Geneva Conference were:

1. The effects of irradiation damage on the physical properties of uranium and uranium-molybdenum alloys, S-3-26 as well as metallographic examinations of irradiated specimens. Although the report concerned the irradiation of uranium, some of the specimens were canned in 18-8 type stainless steel and contained magnesium cores. These materials will have undoubtedly been tested as well.

The data contained in the paper on the properties of uranium and uranium-molybdenum alloys, such as creep, hardness, strength, dimensional changes, are similar to the data obtained in the USA.

2. The effect of irradiation on structural materials S-3-35 was discussed in a paper dealing with the physical property changes of copper, nickel, armco-iron aluminum, stainless steel, carbon steel, and other steel alloys.
It was stated in this paper that the Russian scientists had studied the behavior of lithium metal under neutron irradiation in 1952. This was probably part of their hydrogen bomb study, since one of the products of the lithium-6 isotope when irradiated by neutrons is tritium.

Since these conferences, very little information on irradiation programs has appeared in the literature. A paper by Konobeevsky which presented the theory of post-neutron irradiation damage on uranium and uranium-molybdenum alloy indicates continued fundamental studies on irradiation damage of fuels. S-2-7 Kurchatov at Harwell S-1-1 stated that thorium behaved much better in reactors than uranium, indicating irradiation studies on both of these materials had been conducted. He also mentioned the fact that uranium dioxide (UO2) fuel elements were very stable under irradiation, indicating a program of irradiation studies of UO2. This would be expected, since at the Belgrade Conference S-1-9 it was revealed that UO2 was to be the fuel in the 420 mw, pressurized-water power plants.

As with the program on irradiation effects on UO2, we can conclude from the reactors to be constructed (as listed in the Belgrade report) that irradiation study programs have been or are being conducted on:

1. Liquid metals such as sodium and sodium-potassium eutectic alloy for their proposed liquid metal-cooled experimental power reactors.
2. Zirconium and zirconium alloys for use in all of the power reactors.
3. Steel and other structural materials to be used in their power reactors.

Continuation and expansion of the irradiation effects study program is evidenced by the construction and dedication recently of a new institute for nuclear research at the Academy of Sciences at Dubna, U-33, U-34. This facility will have an experimental reactor operating by the end of 1957 for conducting irradiation experiments.

A new nuclear laboratory has also been established at the USSR Tool Research Institute for studying and utilizing the irradiation effect on the properties of metals and other alloys. U-35

In general, it can be stated the USSR is conducting an extensive program on the study of irradiation effects in fuels, fuel element containing materials, reactor structural materials, and coolant materials, as well as studies on the uses of materials whose properties have been changed by irradiation. Complete information is lacking on the very important question of new test reactor facilities and their capability.

M. Enriched Uranium-235

One interesting fact, the significance of which is not readily apparent, is that there is no mention in the available Soviet literature of the use of highly enriched uranium. A number of the research reactors have used uranium enriched to 10%, but fully enriched uranium has apparently not been used. It is known that the Russians have a diffusion plant ca-
Examination of the data indicated good correlation with that in the USA.

3. The report on the metal research hot laboratory S-3-33 made reference to metallographic examination of uranium as well as the physical property changes of aluminum.

An indication of the excellent "hot" laboratory techniques in the USSR was revealed by the methods used to obtain plastic reproductions of their radioactive metallographic specimens. These slightly radioactive replicas could then be decontaminated readily, aluminized, and examined outside the "hot" cells using an electron microscope and ordinary metallographic techniques.

4. The report on the "hot" chemistry laboratory S-3-34 made reference to their studies on the transuranium elements and the rare-earth fission products.

5. In the Geneva Paper P/662, S/3/19 which was concerned with the study of beryllium and beryllium oxide as a neutron moderator, the author indicated that because of the good nuclear properties of these materials further studies would be made of the physical, mechanical, and corrosion properties of beryllium and beryllium oxide under in-pile conditions.

6. Paper P/683 S-3-36 discussed some of the effects of irradiation on hydrocarbons. In these tests, 75 kv X-rays were used as the radiation source indicating interest in the gamma radiation effect on hydrocarbons.

7. Paper P/673 S-3-32 presented experimental information on the effect of irradiation on water. Excellent data were presented on the rate of radiolysis of water irradiated with fast neutrons and gamma radiation at elevated temperatures with and without the addition of contaminants.

This paper was a good example of the thoroughness of some of the fundamental studies made by the scientists in the USSR.

8. At the Conference of the Academy of Sciences of the USSR on the Peaceful Uses of Atomic Energy, a paper was presented on the changes in reactor-grade graphite under irradiation by neutrons. S-4-16. The information contained on the irradiation behavior of this material was detailed and accurate and compared favorably with the USA information on reactor-grade graphite irradiation.

There is no doubt that extensive irradiation studies have been made and are continuing to be made on this material in order to understand and eliminate or compensate for the changes which occur.

9. At the same conference as in (8) above, another paper was presented on the effect of irradiation on the diffusion of silver in lithium. S-4-17. The experiments were performed as a means of establishing some of the fundamental processes that occur in the changes of the lattice of lithium under irradiation.
pable of producing uranium of some enrichment, and they have enriched to at least 10% U-235. S-1-2 The future Soviet program implies the use of highly enriched uranium in two instances. The experimental homogeneous power reactor would require highly enriched uranium if it is to use thorium as the fertile material, and the PWR type of power station may operate with highly enriched uranium with thorium as the fertile material.

IX. Possible Significant Developments

There are a number of possibilities as to developments which may be going on in the USSR program and which could, if they were successful, be generally recognized as areas in which the Soviet technology surpassed that of the United States. All of these possibilities are very much conjectural, and are based on possible interpretations of certain statements and remarks in the Soviet literature. There is no real evidence that these developments will materialize. Nevertheless, it does not seem reasonable to assume that their program is entirely similar to that of the United States, and it could easily be true that certain developments are under way which are not being talked about. Such developments may or may not be the ones described below.

A High-Temperature, Water- and Steam-Cooled Reactor

This is perhaps the most probable of the various possible developments that are discussed here. It has been stated that the coolant outlet temperature from the graphite-moderated, water-cooled reactor (Power Station No. 2) is 540° C. Other statements of the current program have given lower values for this temperature, but the discussion of the program would appear to substantiate the higher value. The only two methods of attaining high coolant temperatures which have been seriously pursued in the United States are the use of liquid metal coolants and the use of gas coolants. The Soviet reactor, if it attains the reported coolant temperature, must use steam as the coolant, and in this sense it is a gas-cooled reactor. However, it also appears that part of the reactor heat—probably the major fraction—is removed by liquid water at a lower temperature. It is possible that this combination of the high-temperature capabilities of steam with the good heat removal characteristics of water might result in a reactor which, at least thermally, would appear very attractive and perhaps appear to be superior to reactors being built in the United States. The question of whether such a reactor would actually be an attractive one could be answered only when the degree to which the nuclear performance may have been degraded in order to achieve the high
thermal performance is known. It does not appear probable that a reactor of this type with good neutron economy will be developed in the near future. The possibility, however, cannot be ruled out.

**A Cheap Power Reactor for Export**

In one of the discussions of the Soviet program, it was mentioned that a possible reactor type to be developed is a cheap, pressurized water reactor using aluminum fuel element jackets, and operating at relatively low temperature and relatively poor thermal efficiency. Such a reactor would presumably be used for remote locations in the USSR, and, it was stated, could be exported to underdeveloped countries. The United States does not have a program aimed at the development of a very cheap, low performance reactor, although certain experimental reactors such as the Bonar reactors have demonstrated that relatively cheap reactors can be built. It would seem possible that the development of a reactor which could be sold abroad at a price very significantly below that of other reactors might be a worthwhile propaganda achievement, even if the performance of such reactors were not particularly attractive. There is a reasonable probability that such a program, centered around a pressurized water or a boiling water reactor, could be successful.

**Plutonium Recycle**

The recycling of plutonium produced in a nuclear reactor, as a means of enriching natural uranium fuel, is a possibility which is followed with very great interest by all countries which do not have large and efficient isotope separation plants. Any country which can demonstrate the economical use of plutonium in this way will generally be considered a leader in nuclear power technology. The recycling of plutonium is referred to frequently in the Soviet literature, and there is little doubt that it is a part of their program. It is also a part of the British and United States programs. It can be predicted that economic plutonium recycle will develop as a technology, and will not be the result of a single invention. Furthermore, it cannot be demonstrated except through the manufacture of many plutonium fuel elements and the use of those elements in power reactors. Since there is no evidence that the Soviet program has reached such a scale, and indeed, there is not a large power reactor yet to use such fuel elements, it is probably safe to assume that this is not a development which can be claimed by the Soviet Union in the next few years. It is, however, one which will no doubt be pushed vigorously and in which the Soviet Union could conceivably become a leader over a period of years. It may well be that Britain rather than the United States will be the major competitor in this field, since the British have a real economic incentive for plutonium recycle.

**Beryllium as a Reactor Material**

Quite evidently the Soviets have done a very considerable amount of development work on beryllium. This material is apparently also being developed intensively in England, and work is also being done in the United States. It appears at this time that beryllium may be a necessity for
high-temperature gas-cooled reactors of good neutron economy, and a nation which developed practical techniques for the cheap fabrication and use of this material would have a decided advantage in the gas-cooled reactor field. There is no specific evidence to indicate that the USSR is ahead in such a development.

Undermoderated H2O Reactors

A good deal of theoretical attention has been given by the Soviets to reactors which are cooled and moderated by water, but in which the ratio of uranium to water is rather high. Such reactors might have quite good neutron economy, but would require high enrichments. This concept is apparently not being pushed for the pressurized water power stations. Although the reactor type may have some advantages, it is not likely to give an important advantage to the country which develops it.

Nuclear Propulsion

If the Soviets are in a position to build the pressurized water power stations on something like their stated schedule, then it is quite probable that they have the technology for building naval reactors of essentially the same type as the successful naval reactors in the United States. It is probable that this reactor type will be the most suitable for naval propulsion over the near future. When such reactors are available, the ship performance which can be obtained with them is a function not only of the excellence of the reactor design, but also of the technical skill in designing the installation, and the degree of advancement in the more conventional parts of the propulsion machinery. It is our impression that USA naval practice maintains a rather conventional attitude toward the non-nuclear parts of a nuclear propelled ship. It is quite conceivable that the Soviets, by taking a bolder approach in these areas, could develop naval vessels of higher performance (e.g. speed), than those of the United States. This would not, of course, necessarily mean that the vessels were superior as all-round weapons.

In the field of aircraft propulsion, it is necessary to attain quite high reactor coolant temperatures. The most plausible reactor types are direct-cycle air-cooled types and types using liquid metals or molten salts as the primary heat transport fluid. There is no direct evidence of large programs in either of these directions in the USSR (except, of course, the fast reactor and sodium-graphite projects). In view of the Soviet success in the missile field, there would seem to be some question as to their need for a nuclear-powered aircraft. It would not be at all surprising, however, if the Soviets were experimenting in the direction of aircraft nuclear reactors. Indeed, it seems very probable that some work of this type is going on. The technology for such an application would not be very closely related to the power reactor technology, and no guesses can be made on the basis of information in the unclassified power reactor field.
The recent announcement of Soviet plans to build a nuclear locomotive cannot be dismissed as technically unfeasible. Such a device would be technically quite feasible in the United States, but for economic and safety reasons the development has not been pursued here. It may be that there is a real use for a nuclear locomotive in the USSR. If so, the successful construction of one can be anticipated. The possession of such a device, however, would not seem to represent any great technical or economic advantage for the USSR.
APPENDIX I

Calculations

1. Plant Factor

\[ \frac{7500 \text{ hours/yr}}{365 \text{ days/yr} \times 24 \text{ hrs/day}} = 0.85 \]

2. Specific Power

a) For FWR: \( \frac{1520 \text{ MW}}{80 \text{ T}} = 19 \text{ MW/ton of fuel} \)

b) For APS: \( \frac{1150 \text{ MW}}{370 \text{ T}} = 3.11 \text{ MW/ton of fuel} \)

3. Annual Burnup of Fissionable Material

\[ 1.3 \text{ grams/MWD} \times \left[ \frac{\text{Heat Power in MW}}{1 \text{ MW/T}} \right] \times 365 \text{ days} \times 0.85 \]

a) For FWR: \( 1.3 \times 1520 \times 365 \times 0.85 = 610 \times 10^3 \text{ grms annual burnup} \)

b) For APS: \( 1.3 \times 1150 \times 365 \times 0.85 = 462 \times 10^3 \text{ grms annual burnup} \)

4. Fuel Lifetime

a) For FWR: \( \frac{3500 \text{ MWD/T}}{19 \text{ MW/T} \times 0.85} = 209 \text{ days or } \sim 7 \text{ months} \)

b) For APS: \( \frac{2500 \text{ MWD/T}}{3.11 \text{ MWD/T} \times 0.85} = 944 \text{ days or } \sim 2.6 \text{ years} \)

5. Average Enrichment of FWR

\[ \frac{17 \text{ tons} 	imes 0.00714 + 23 \text{ tons} 	imes 0.015}{40} = 0.0117 \]

6. \text{\textsuperscript{235}U Content}

a) For FWR: 40 tons oxide fuel \( \times 0.88 = 35.2 \text{ tons uranium metal} \)

b) \( 35.2 \times 0.0117 = 0.41 \text{ tons or } 410 \text{ kg \text{\textsuperscript{235}}U per reactor} \)

b) For APS: 185 tons metal fuel \( \times 0.012 = 2.22 \text{ tons or } 2220 \text{ kg \text{\textsuperscript{235}}U per reactor} \)
7. Total Pu Production per Fuel Lifetime
   a) For FWR: \[ 410 \text{ kg } \text{U}^{235} \times 0.28 \times \frac{\text{gms Pu}}{\text{gms U}^{235}} = 115 \text{ kg Pu per fuel lifetime per reactor.} \]
   b) For APS: \[ 2220 \text{ gms U}^{235} \times 0.135 \times \frac{\text{gms Pu}}{\text{gms U}^{235}} = 286 \text{ kg Pu per fuel lifetime per reactor.} \]

8. Annual Pu Production
   a) For FWR: \[ 115 \text{ kg Pu} \times \frac{365 \text{ days}}{209 \text{ days}} = 200 \text{ kg Pu/year per reactor.} \]
   b) For APS: \[ 2220 \text{ kg Pu} \times \frac{365 \text{ days}}{244 \text{ days}} = 111 \text{ kg Pu/year per reactor.} \]

9. Projected Annual Consumption and Production of Fissionable Material
   Assume: 1.5% enrichment; 99% efficiency; and plant factor of 0.85.
   a) \[ = \text{fraction U}^{235} \text{ burned} \times \frac{1000}{\text{MWD/T}} \times \frac{\text{eMW}}{0.29} \times 365 \times 0.85 \times 0.015 \]
      \[ = 16,000 \times \text{fraction U}^{235} \text{ burned} \times \frac{\text{eMW}}{\text{MWD/T}} \]
   b) Annual Plutonium Production**
      \[ = \text{annual consumption of U}^{235} \times 0.85 \]
   c) Maximum available production of plutonium, assuming that the fuel remains in the reactor a short time so that effectively none of the plutonium produced is burned:
      \[ = \left[ 1.3 \text{ gms/MWD} \times \frac{\text{eMW}}{0.29} \times 365 \times 0.85 \right] \times 0.85 \]

10. Annual Requirement of Replacement Fuel
    \[ = \frac{365 \times 0.85}{0.29} \times \frac{\text{eMW}}{\text{MWD/T}} = 1070 \times \frac{\text{eMW}}{\text{MWD/T}} \]

** This production would not be available until approximately a year, to account for the time the fuel is in the reactor and in the processing plant.
APPENDIX II

Calculations to Estimate Physics Characteristics of the Soviet PWR

Core Lattice Dimensions (Reference S-1-11) Taken in Calculations

- Core diameter = 3 m (9.84 ft)
- Core height = 2.5 m (8.22 ft)
- 349 hexagonal casings with fuel elements
- 91 fuel rods per casing (approximately)
- Fuel rod diameter = 10.2 mm (0.405 inch)
- Pitch of fuel rods = 14.3 mm (0.563 inch)
- Triangular lattice of fuel rods
- Zirconium clad thickness around fuel rods = 0.030 inch (assumed)

Average Concentrations of Materials in Core

Area of dashed triangle = \( \frac{1}{2} \times 0.563 \times 0.563 \)
\[ = 0.1373 \text{ in}^2 \]

Area of \( \frac{1}{3} \) of a tube = \( \frac{\pi \times 0.405^2}{3} \)
\[ = 0.0645 \text{ in}^2 \]

Area of water within dashed triangle = \( 0.1373 - 0.0645 \)
\[ = 0.0728 \text{ in}^2 \]

Hence, \( \frac{\text{volume water}}{\text{volume fuel + clad}} = \frac{0.0728}{0.0645} = 1.13 \)

\[ \frac{\text{volume fuel + clad}}{\text{volume fuel}} = \left( \frac{0.405}{0.405 - 2 \times 0.30} \right)^2 = 1.38 \]

Hence, water to oxide volume ratio = \( 1.38 \times 1.13 = 1.56 \)

\[ \frac{\text{volume Zr-clad}}{\text{volume fuel}} = 1.38 - 1 = 0.38 \]. Average water temperature = 262.5°C
\[ = 505^\circ \text{F} \]

Water pressure = 100 atm = 1470 lb/in² abs. From Steam Tables, water density = 49.4 lb/cu ft = 0.793 gm/cc.

Average fuel enrichment based on quoted 17 metric tons of natural uranium dioxide and 23 tons of 1.5 percent enriched uranium dioxide is:
\[ \epsilon = \frac{(17 \times 0.714) + (23 \times 1.5)}{17 + 23} = 1.17 \text{ percent} \]

Density of UO₂ fuel = 10.5 gm/cc assumed
Check on total UO₂ - tonnage (metric):  
\[ \text{Tons} = \frac{\pi \times (0.405 - 0.060)^2}{4 \times 144} \times \frac{7.5 \times 91 \times 349 \times (10.5 \times 62.4)}{2200} \]  

Tons = 166 compared to 100 tons quoted.

In above calculation, 7.5 feet is taken to be the active length of the fuel rods (8.22 ft. total length). The discrepancy of 6 tons may be that lower density uranium-oxide than 10.5 gm/cc is used. For the physics calculations, however, the high-density value is used inasmuch as it is believed that the Russians will be able to produce the higher density oxide.

Cross Sections

For the core operating temperature, \( E_{th} = 0.0462 \text{ ev.} \) is the energy of the thermal neutrons corresponding to their most probable velocity. At the energy, from BNL-325 using the \( 1/v \) law for absorption cross sections:

\[
\begin{align*}
\sigma_a (\text{Zr}) &= 0.133 \text{ barns} \\
\sigma_a (\text{H}) &= 0.245 \\
\sigma_a (\text{U}^{235}) &= 495.0 \\
\sigma_a (\text{U}^{238}) &= 2.02 \\
\sigma_f (\text{U}^{235}) &= 418 \\
\sigma_a (\text{U}) &= 0 \\
\sigma_a (\text{U}^{238}) &= 0 \\
\sigma_a (\text{U}) &= \frac{0.0117}{1.0117} \times 495 + \frac{1}{1.0117} \times 2.02 = 7.74 \text{ barns} \\
\sigma_f (\text{U}) &= \frac{0.0117}{1.0117} \times 418 = 4.84 \text{ barns} \\
\end{align*}
\]

Regeneration factor, \( \eta = 2.47 \frac{4.84}{7.74} = 1.545 \)  

where 2.47 is neutrons produced per fission process.

In the above, subscript "\( a \)" refers to an absorption process and "\( f \)" to the fission process.

\( N \) is the number of nuclei per cm³ of the material and is given by  
\[
\text{density (gm/cm³)} \times 6.023 \times 10^{23} \times \text{atomic weight number}
\]

The macroscopic absorption cross section is given by \( \Sigma_a = N \sigma_a \).
Calculation gives:

\[ N^{24}_{\text{U}} = 0.0425 \times 10^{24} \]
\[ N^{32}_{\text{O}} = 0.0266 \times 10^{24} \] (To obtain molecules/cm³ of material, molecular weight used in place of atomic weight)
\[ N^{16}_{\text{H}} = 0.0332 \times 10^{24} \] (2 atoms of H per molecule H₂O)
\[ N^{19}_{\text{O}} = 0.0234 \times 10^{24} \]
\[ N^{19}_{\text{F}} = 0.0234 \times 10^{24} \]

Hence, \( \Sigma_a (\text{U}) = 0.0234 \times 10^{24} \times 7.74 \times 10^{-24} \times \frac{1}{1.128} = 0.1605 \text{ cm}^{-1} \)

where \( \frac{1}{1.128} \) accounts for the effect of a Maxwellian distribution of thermal neutrons on absorption processes for 1/\( v \)-absorbers. (Glasstone and Edlund) U-37

Similarly,

\[ \Sigma_a (2\text{r}) = 0.00502 \text{ cm}^{-1} \]
\[ \Sigma_a (\text{H}) = 0.0116 \]
\[ \Sigma_a (0) = 0 \]

\( ^{235} \text{U} \) Fuel Loading

Tons oxide = 46 from above

Tons U = 0.88 \times 46 = 40.5

Kg U = 1000 \times 40.5 = 40,500

Kg \( ^{235} \text{U} \approx 0.0117 \times 40,500 = 475 \)

Thermal Neutron Flux

Total nuclei \( ^{235} \text{U} = \frac{475,000 \times 6.023 \times 10^{23}}{235} = 1.2 \times 10^{27} \)

Total fissions/sec = \( 3.12 \times 10^{10} \times 760 \times 10^6 = 2.4 \times 10^{19} \)

where 760 = heat power of reactor in Mw and 3.12 \( \times 10^{10} \) is fissions/sec/watt of reactor power.

Average thermal flux, \( \bar{\phi} = \frac{2.4 \times 10^{19}}{1.2 \times 10^{27} \times \frac{418}{1.128} \times 10^{-24}} \)

\[ = 5.4 \times 10^{13} \text{ neutrons/cm}^2 \text{ sec} \]
Equilibrium Xe$^{135}$ Concentration

From Glasston and Edlund, U-37

$$\Sigma(\text{Xe}) = \frac{\sigma_x (\gamma_I + \gamma_x)}{\lambda_x + \sigma_x \Phi}$$

where $\lambda_x = 2.1 \times 10^{-5}$ sec$^{-1}$

$$\Phi = 5.4 \times 10^{13}$$

$$\gamma_I + \gamma_x = 0.059$$

$F = \text{fissions/sec cm}^3 \text{ fuel slug}$

$$\sigma_x = 3 \times 10^6 \times 10^{-24} \text{ cm}^2/\text{nuclei}$$

Total cm$^3$ fuel in reactor = $\frac{\pi x 0.3}{4} \times 7.5 \times 91 \times 349 \times 30.5^3 = 1.3 \times 10^6$

Hence, $F = \frac{2.14 \times 10^{-19}}{1.3 \times 10^6} = 5.5 \times 10^{-12}$

Then $\Sigma(\text{Xe}) = 0.006 \text{ cm}^{-1}$ over fuel slug volume only.

Samarium$^{149}$ Absorption

From Glasston and Edlund, U-37

$$\Sigma(\text{Sm}) = \frac{0.0114 F}{4.1 \times 10^{-6}} \times 5.3 \times 10^{-20} = 0.001 \text{ cm}^{-1} \text{ over fuel slug.}$$

Thermal Utilization

$$f^* = \frac{V_{\text{fuel}} \Sigma_a U}{V_{\text{H}_2\text{O}} \Sigma_a \text{H}_2\text{O} + V_{\text{Zr}} \Sigma_a \text{Zr} + V_{\text{poisons}} \Sigma_a P + V_{\text{fuel}} \Sigma_a U}$$

where $V$'s are relative volumes.

$$V_{\text{fuel}} \sim \frac{\pi x 0.3}{4} = 0.935$$

$$V_{\text{Zr}} \sim \frac{\pi x 0.105}{4} = 0.0935 \times 0.0355$$

$$V_{\text{poisons}} \sim 0.0935$$

$$V_{\text{H}_2\text{O}} = 1.56 \times 0.0935 = 0.146$$

$$f^* = \frac{0.0935 \times 0.1605}{0.146 \times 0.0916 + 0.0355 \times 0.005 + 0.0935 \times 0.007 + 0.0935 \times 0.1605} = 0.855$$

*To correct for disadvantage factors, adjust downward by multiplying by 0.98, as found in experiments of WAPD-T-572, U-38 to give $f = 0.837$.
Resonance Escape Probability and Fast Fission Effect

These are based on experimental results presented in WAPD-T-572. U-38

We have $W/U_2 = 1.56 \times 0.793 = 1.23$ (Effective water to uranium-oxide
volume ratio)

where 0.793 is the water density in the core ($\text{gm/cm}^3$).

Effective $W/U = \frac{18.7 \times 270}{10.5 \times 238} \times 1.23 \approx 2.4$ (from WAPD-T-572)

From page 42, Figure 8, of WAPD-T-572, $p \approx 0.75$

From Figure 12, $\delta_{28} = 0.095$

and $e = 1 - \frac{1.5}{2.2} \times 0.095 = 0.057$ so $e = 1.057$

Here, we have taken $\nu'_{25} = \nu'_{28} = 2.5$

Infinite Multiplication, $k_{\infty}$

$k_{\infty} = \eta \text{ epf} = 1.545 \times 1.057 \times 0.75 \times 0.837 = 1.025$

Effective Multiplication, $k$

The migration area, $M^2$, for fission neutrons is estimated, on the basis of past experience, to be 66 cm$^2$. This number is good to within about 15 percent, which is more than adequate for the present case.

Effective core height $H_c = (8.22 \times 30.5 + 2 \times 8) = 266$ cm

Effective core radius $R_c = \left( \frac{2 \times 81}{2} \times 30.5 + 8 \right) = 158$ cm

where the reflector radius has been taken equal to 8 cm for plain water.

Then geometric buckling $B^2 = \frac{\pi}{266} + \frac{2 \times 0.05}{158} = 3.71 \times 10^{-4}$

$1 + N^2 B^2 = 1.025$

$k_{\text{eff}} = \frac{k_{\infty}}{1 + N^2 B^2} = 1.00$

This is the effective multiplication calculated for the reactor at operating temperature with Xe$^{135}$ and Sm$^{149}$ poisoning effects included. The result indicates that no reactivity is available for operation. However, the calculation has been done for the case of a single average enrichment of 1.17 percent rather than a two-zone reactor. If the 1.5% enriched fuel were placed in a central zone and the natural uranium as a blanket surrounding the central zone, then slightly greater reactivity than calculated by the above method would be obtainable. This extra activity could amount to the order of 1 - 1.5% giving $k_{\text{eff}} = 1.01$ to 1.015. For the
relatively short cycle lifetime planned and the high conversion ratio expected, this value of keff is probably just slightly smaller than needed so that a very slight increase in average enrichment, say to 1.25 percent, would probably be adequate. At any rate, the uncertainties in such calculations are of the same order of magnitude as the effects introduced by this difference in enrichment.

**Initial Conversion Ratio, I.C.R.**

\[
I.C.R. = \frac{\sigma_a (U^{238})}{\sigma_a (U^{235})} + \frac{\eta_{235} e^{(1-p)}}{1 + b^2}
\]

\[
= \frac{2.02}{0.0117 \times 1.95} + \frac{2.07 \times 1.057 \times 0.25}{1.025}
\]

\[
= 0.35 + 0.53 = 0.88 \text{ (ave)}
\]

For the 1.5% enriched fuel in a central zone and the natural uranium in a blanket zone around the enriched fuel, the I.C.R. would be slightly less than calculated. Also, some control-rod absorptions occur (although for the keff calculated, the rods would be nearly out of the core) which further reduce the I.C.R. from the value calculated. However, the reductions due to these two effects should be less than 10 percent so that

\[
I.C.R. > 0.80 \quad \text{should be obtainable.}
\]

**Plutonium Production Based on Numbers used in Physics Calculations**

From curves on the long-term reactivity changes, U^{235} burnup, and Pu production in a U^{235}-U^{238}-Pu system, the following results are obtained for the quoted case of 3500 Mw-days of operation per metric ton of uranium fuel in the reactor:

- Initial Kg U^{235} in Reactor = 475
- Kg U^{235} left in reactor after 3500 (MwD/T) of operation = 316
- Kg U^{235} burnup/cycle = 159
- Kg Pu in Discharged Fuel = 136
- \[\frac{k_{\text{eff}} \text{ at end of cycle}}{k_{\text{eff}} \text{ initial}} \approx 0.99\]

**Plutonium Recycle**

Figure 3 of report HW-507001-U-39 gives the minimum conversion ratio requirement to achieve plutonium recycle with natural uranium feed for different initial U^{235} enrichments. From this figure, for 1.17 percent average enrichment in the Russian PWR, the minimum conversion ratio re-
required to achieve plutonium recycle is obtained as about 0.62. For a two-
zone fuel distribution, the minimum conversion ratio could probably be as
high as 0.70. Hence, for an I.C.R. value of > 0.80 for the Russian PWR,
plutonium recycle is possible.
APPENDIX III

Calculations on Graphite-Moderated, Water-Cooled Reactors

1. Heat Transfer Surface

There are 128 fuel assemblies, each containing 4 fuel tubes. Each fuel tube has O.D. = 9 mm and wall thickness = 0.1 mm. Length is 1.7 m. S-2-2 I.D. = 9 - 2 (0.4) = 8.2 mm.

Heat transfer surface = \( \frac{(\pi)(0.82)}{100} \times 1.7 \times 128 \times 4 = 22.4 \text{ m}^2 \)

Total heat rate = \( 30,000 \times \frac{8.6 \times 10^5}{10^3} = 25.8 \times 10^5 \text{ k cal/hr} \)

average heat flux = \( \frac{25.8 \times 10^5}{22.4} = 1.15 \times 10^6 \text{ k cal/m}^2 \text{ hr} \)

max = 1.6 (by Reference S-2-2)

av

maximum heat flux = 1.15 \( \times 10^6 \times 1.6 = 1.84 \times 10^6 \text{ k cal/m}^2 \text{ hr} \)

This checks with statements in Reference S-2-3.

2. Uranium Thickness

Total uranium loading = 550 kg (Reference S-2-2)

Loading per tube = \( \frac{550}{(128)(4)} = 1.095 \text{ kg} \)

Loading per cm of tube = \( \frac{1095}{170} = 6.41 \text{ gm} \)

Volume per cm = \( \frac{6.41}{18.9} = 0.341 \text{ cm}^3 \)

0.341 = \( \pi (R^2 - 0.41^2) = \pi (R^2 - 0.168); \quad R^2 = \frac{0.341}{\pi} + 0.168 \)

\( R^2 = 0.1085 + 0.168 = 0.277; \quad R = 0.526 \text{ cm} \)

Thickness = 0.526 - 0.41 = 0.116 cm

3. Size of Reactor of This Type for Ship Propulsion

Reference S-2-3 states the core diameter of the APS reactor is 1.5 meters (output = 30,000 kW of heat). References U-14 and S-1-7 states that the output of the icebreaker is 200 MW of heat.
Core volume for icebreaker application = \( \frac{200}{30} = 6.67 \)

Core volume of APS reactor

Diameter for icebreaker application = \( 3\sqrt{6.67} \) (115)

= 2.82 m

= 9.25 ft.

For submarines or freighters, half this output would be ample, and core diameter would be

\( \approx 3\sqrt{0.5} \) (9.25) = 7.3 ft.

4. Thermal Utilization

From Reference S-2-1, \( k_{\infty} = 1.394 \); \( p = 0.92 \)

For 5% enrichment:

\( \eta = \frac{(2.46)(0.05)(580)}{(0.95)(2.75)+(0.05)(687)} = 1.93 \)

\( f = \frac{k}{\eta p} = \frac{1.394}{(1.93)(0.92)} = 0.785; \quad \frac{1}{f} = 1.274 \)

This checks fairly well with table (page 106) of Reference S-2-1.

5. Flow Velocity in Fuel Tubes

Reference S-2-2 gives the primary flow rate as 240 tons/hr (probably metric tons).

\( 240 \text{ tons/hr} = \frac{240 \times 10^6}{3600} = 0.667 \times 10^5 \text{ cm}^3/\text{sec} \)

Flow area = \( \pi(0.41)^2(502) = 265 \text{ cm}^2 \)

Flow velocity = \( \frac{667 \times 10^2}{265} = 252 \text{ cm/sec} \)

= 8.3 ft/sec (av.)

Assume the average velocity in the downflow tubes is the same, then the I.D. of the downflow tubes is:

\( \text{I.D.} = 0.82\sqrt{4} = 1.64 \text{ cm} \)

6. Stress in Tubes

For fuel tubes, \( S = \frac{Pd}{2t} = \frac{(1500)(0.82)}{(2)(0.04)} = 15,000 \text{ psi} \)
Can probably design to 20,000 psi in downflow tube.

\[ t = \frac{PD}{2S} = \frac{(1500)(1.64)}{(2)(20,000)} = 0.06 \text{ cm} \]

7. **Total Steel in Core**

\[ V = \left[ \left( \pi \cdot 0.82 \right) \cdot 0.064 \right] + \pi \cdot 1.05 \cdot 0.02 \cdot 4 + \pi \cdot 0.164 \cdot 0.06 \]

\[ (170)(128) = 21,400 \text{ cm}^3 \]

Weight of steel = \[ 21.4 \times 10^{-3} \times 7.5 = 161 \times 10^3 \text{ gm} \]

\[ \frac{(V \Sigma)_{\text{steel}}}{(V \Sigma)_{u}} = \frac{161}{550} \times \frac{236}{56} \times \frac{3}{37} = 0.101 \]

Assume disadvantage factor of 1.3; let \( A \) designate total absorption, then

\[ \frac{A_{\text{steel}}}{A_u} = (1.01)(1.3) = 0.131 \]

8. **Thermal Utilization**

From Reference S-2-3:

\[ C/u = 171; \quad H/u = 4.2 \]

Assume disadvantage factor of 1.1 for H\(_2\)O, 1.5 for C, then

\[ \frac{A_C}{A_u} = 171 \times \frac{4.4 \times 10^{-3}}{37} \times 1.5 = 0.0305 \]

\[ \frac{A_H}{A_u} = 4.2 \times \frac{0.33}{37} \times 1.1 = 0.011 \]

\[ \frac{1}{f} = 1 + \frac{A_{\text{steel}}}{A_u} + \frac{A_C}{A_u} + \frac{A_H}{A_u} = 1 + 0.131 + 0.031 + 0.011 \]

\[ f = 1.203; \quad f = 0.83 \]

This is larger than the \( f \) calculated for criticality in (4) above.

For this case, \[ \frac{1}{f} = \frac{1}{0.785} = 1.275. \] If we assign the discrepancy to the steel absorption, then

\[ \frac{A_{\text{steel}}}{A_u} = 1.275 - 1 - 0.031 - 0.011 = 0.203 \]
9. Buckling (hot) S-2-3

\[ b^2 = \left( \frac{2.405}{113} \right)^2 + \left( \frac{\pi}{222} \right)^2 = 0.000453 + 0.000200 = 0.000653 \]

\[ \tau = 260 \]

\[ \tau b^2 = (260)(0.000653) = 0.170; \quad e \tau b^2 = 1.185 \]

\[ L_2h^2 = (92)(0.000653) = 0.060 \]

10. Conversion Ratio (hot) (Reference S-2-3) \((p = 0.908\text{ hot})\)

\[ R_c = \frac{0.57}{5} + \frac{2.08}{1.185} = 0.0814 + 0.1614 = 0.243 \]

Alternate Calculation:

\[ R_c = \frac{\pi 25}{e \tau b^2} - R_{xe+sm} - 1 - R_{steel} - R_H - R_c - \left( 1 + \frac{0.57}{5/0.714} \right) L_2 b^2 \]

where \( R_{steel} = A_{steel}/A_{25} \) etc.

\[ \frac{A_{25}}{A_{th}} = \frac{34.3}{36.9} = 0.932 \]

\[ R_{steel} = 0.203/0.932 = 0.213 \]

\[ R_H = 0.0411/0.932 = 0.044 \]

\[ R_c = 0.305/0.932 = 0.333 \]

\[ R_{xe+sm} = 0.05 \text{ (Figure 2, Reference S-2-3)} \]

\[ R_c = \frac{2.06}{1.185} - 0.05 - 0.218 - 0.044 - 0.033 - (1.08)(0.06) \]

\[ = 1.755 - 1.345 - 0.065 = 0.345 \]

Discrepancy is probably due to absorption in control rods. So take

\[ R_{control} = 0.10 \]

11. Possible Improvement in Conversion Ratio

a) If reactor is made large, lattice is kept essentially constant except that \( p \) is increased slightly to soak up the excess neutrons. Then \( \tau b^2 \) might be reduced to, say, 1.06 and \( L_2 b^2 \) to \( \sim 0.02 \); enrichment might go down a little, but assume we keep it nearly the same, then:
\[ R_c = \frac{2.08}{1.06} - 0.05 - 1 - 0.218 - 0.044 - 0.033 - (1.08)(0.02) \]
\[ = 1.962 - 1.345 - 0.0216 = 0.595 \text{ if there is not absorption in control rods, or } \sim 0.5 \text{ if we lose 0.1 to control.} \]

However, the steel absorption may be overestimated by 0.077 (Section 8 above), so possibly \( R_c \) approaches 0.6 even with control rods.

b) If, in a large reactor, "\( R_{steel} \) can be reduced, say, to 0.03 by replacing steel by zirconium, then

\[ R_c = \frac{2.08}{1.06} - 0.05 - 1 - 0.03 - 0.044 - 0.033 - 0.022 = 0.783 \]

without losses to control rods, or perhaps 0.7 to 0.75 with rods. However, enrichment must be maintained at ~3%.

c) If, in a large zirconium reactor, enrichment is dropped to 2.5%, then all parasitic absorptions (except Kx and Sm) are multiplied by 2, and conversion ratio becomes approximately:

\[ R_c = \frac{2.08}{1.06} - 0.05 - 1 - 0.06 - 0.088 - 0.066 - 0.022 = 0.68 \]

without losses to control rods.

d) If the resonance escape were held constant in the large reactor and enrichment were allowed to drop, then:

\[ k_{\infty} (\text{crit.}) = (1.06)(1.02) = 1.08 \]
\[ k_{\infty} (\text{required}) = 1.08 (1.092) = 1.180 \]
\[ \eta f = \frac{1.180}{0.908} = 1.30 \]

Try 3% enrichment:

\[ \eta = \frac{(2.46)(0.03)(580)}{(0.97)(2.75)+(0.03)(687)} = \frac{12.8}{2.67 + 20.61} = 1.838 \]

\[ \frac{1}{f} = 1 + 0.274 \times \frac{71.44}{42.8} = 1.457; \quad f = 0.687 \]

\[ \eta f = (1.838)(0.687) = 1.261. \text{ This is about right. (Misses by ~3%) } \]

Conversion ratio then is:

\[ R_c \sim \frac{0.57 + 2.08 (1 - 0.908)}{1.02} = \frac{0.136 + 0.188}{0.714} = 0.322 \]

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c) If amount of U could be doubled without increasing amount of steel appreciably, then $R_c$ might increase by about 0.1, and be $\sim 0.4$ or a bit better for the large reactor with $\sim 3\%$ enrichment.
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Comments on CIA/61 62-26, A Technical Evaluation of the
Soviet Nuclear Power Program, 15 May 63, S.

1. The preface of this report, prepared under CIA/OSI external research contract by the [redacted], outlines the objectives as follows:

"The objectives of the report are to estimate the technical feasibility of the Soviet nuclear power program as outlined in the Sixth Five-Year Plan (1956-60) and the probable course of this program in the years 1961-70. A secondary objective of the project is to estimate the effect of the Soviet nuclear power program on their stockpile of fissionable materials."

It is further stated that:

"The estimates presented herein are not contrary to the immediate views of the Office of Scientific Intelligence."

2. Our comments apply only to what appear to be erroneous estimates presented in the report concerning the total installed electric generating capacity of the USSR. Since these estimates were instrumental in deriving estimates of the probable course of the Soviet nuclear power program in the years 1961-1970, one of the main objectives of the report, corrections of this data may affect the conclusions of the report or at least "set the record straight." The report states:

"According to the reply of the USSR to the US questionnaire, (6-1-8) there will be a total of about 23,300 MW electrical generating capacity in 1960. The nuclear power program will therefore constitute about 3.7% of the total Russian generating capacity."

On page 12, however, 23,300 MW is used as the capacity for 1957, and 43,000 MW for 1960. Moreover, in getting the 3.7%, the author divides what is apparently a nuclear power estimate of 990 MW by 23,300 MW

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representing total capacity — but the 990 MW estimate has not been explicitly made up to this point and is never made anywhere in the paper. 1,775 MW is given as the apparent plan for 1960 in Table I, page 7, 715 MW is given in Table III, page 15, as the estimate of average capacity during 1960 and Figure I shows a maximum of 900 MW and a minimum of 600 MW for 1960.

According to the original Sixth Five-Year Plan, total electric generating capacity at the end of 1960 is planned to be 2.2 times that at the end of 1955. Since the Soviets had 37,450 MW installed at the end of 1955, this would mean that 81,800 MW were planned for the end of 1960. There are indications, however, that this goal will not be fulfilled. Currently, we are estimating that only about 65,000 MW will be installed by that time. This is more than double the 26,300 MW figure applied to 1960 and 50% greater than the 43,000 MW figure applied to 1960 on page 12. The figure of 990 MW works out to be about 1.5% of our estimate of Soviet electric generating capacity at the end of 1960 (65,000 MW), or 1.2% of the original Soviet goal (81,800 MW).

3. The report states:

"The published expected total power production in the USSR (8-1-8) for the years 1957 (211 x 10^{-9} kwh or 26,300 MW capacity) and 1960 (320 x 10^{-9} kwh or 43,000 MW capacity) are projected to the year 1970 (Figure 1) the anticipated total power capacity in Russia would be about 140,000 MW."

The above production figures are reasonably correct, but the capacity figures are considerably lower than we would estimate. In addition, the production figures were divided by an average annual hours of operation of about 7,500 hours to derive the capacity estimates of 26,300 MW and 43,000 MW. The Soviets, it is true, have claimed that nuclear power plants will operate 7,500 hours per year as base
load plants, but this figure is unrealistic as an average for the USSR. The average in 1955 was 4,955 hours, decreasing somewhat in 1956 and 1957.

In Figure 1, the estimate of 140,000 MW in 1970 was derived by projecting the annual rate of growth of increase of capacity from 1957 to 1960. Based on the erroneous figures of 23,300 MW and 43,000 MW such a simple projection would give 120,000 MW in 1970 instead of the 140,000 MW as shown in the graph and as stated on pages 12 and 13.

If the correct figures for 1957 were used (47,000 MW) in conjunction with the planned capacity figure corresponding to the planned 1960 production of 320 billion kWh (81,800 MW), then a projection of this rate of growth would give the absurd figure of 480,000 MW in 1970.

Our current estimate of total electric generating capacity at the end of 1970 is about 160,000 MW, which assumes a reduction in the rate of growth. Thus, the 3 errors in the report were compensating so that the 1970 figure derived from faulty methodology and faulty data happens to be near our current estimate. (Errors in 1) estimating 1957 and 1960 capacity, 2) methodology of projecting annual growth through 1970, and 3) apparent mechanical error in projection.

p. 13  "Assuming that by 1970 the nuclear power capacity will quite likely have reached 5-10% of the total power generating capacity (estimated USA value is 6.3%; UK is about 3%), then an installed nuclear capacity of 7,000 to 14,000 MW may be expected."

Evidently the estimates of the Soviet nuclear power program through 1970 are based upon an arbitrarily assumed relation of nuclear power plant capacity to total electric generating capacity in 1970. Through a series of compensating errors the figure of 140,000 MW electric generating capacity in 1970 is within reason. However, if our estimate of 160,000 MW is used,
the nuclear capacity in 1970 derived from the above 5-10% would be 8,000 to 16,000 MW.