-6909

ANL-6909

Argonne National Laboratory

CATALOG OF NUCLEAR REACTOR CONCEPTS

Part I. Homogeneous and
Quasi-Homogeneous Reactors

Section II. Reactors Fueled with Homogeneous
Aqueous Solutions and Slurries

by

Charles E. Teeter, James A. Lecky,
and John H. Martens

Price $2.75 . *Available from the Office of Technical Services, Department of Commerce, Washington 25, D.C.*

ANL-6909
Reactor Technology
(TID-4500, 32nd Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60440

CATALOG OF NUCLEAR REACTOR CONCEPTS

Part I. Homogeneous and Quasi-Homogeneous Reactors
Section II. Reactors Fueled with Homogeneous Aqueous Solutions and Slurries

by

Charles E. Teeter, James A. Lecky,
and John H. Martens

Technical Publications Department

July 1964

Operated by The University of Chicago
under
Contract W-31-109-eng-38
with the
U. S. Atomic Energy Commission

TABLE OF CONTENTS

PREFACE

This report continues the catalog of concepts for nuclear reactors that was begun in ANL-6892. As in the previous report, the material is divided into chapters, each with text and references, plus data sheets that cover the individual concepts.

J.H.M.
July 1, 1964

PART I HOMOGENEOUS AND QUASI-HOMOGENEOUS REACTORS

SECTION II REACTORS FUELED WITH HOMOGENEOUS AQUEOUS SOLUTIONS AND SLURRIES

Chapter 1 Introduction

The use of aqueous solutions and slurries to provide an even dispersion of fuel atoms among moderator atoms was one of the earliest suggestions for nuclear reactors.[1] The concepts in this section follow the early idea and are thus distinguished by a common characteristic: fuel dissolved or suspended to form a homogeneous aqueous solution, colloidal dispersion, or slurry.

These concepts have been investigated by many workers. Especially valuable as sources of detailed information are reviews by Lane et al.[1,2] For engineering details of specific reactors, see the tabulations referred to in the General Introduction to the catalog.

The title of the section defines generally how concepts have been chosen for inclusion, but some explanation may clarify why certain related concepts were excluded.

The term "homogeneous aqueous" would strictly mean that the core must be an aqueous solution or suspension throughout, with no solid moderator. A few concepts in which the fuel solution is in contact with such a moderator are included here, however, because of their small number and their close similarity to the other concepts included. "Aqueous" includes light water, heavy water, and any water-containing solution or suspension. Molten salts and liquid-metal solutions or slurries are discussed in Sections III and IV respectively. Solutions and suspensions are here defined to include particle sizes from those of true solutions to particles that can be made into a slurry--up to 500 mμ in diameter. Aqueous suspensions and slurries are included in this section because their properties and behavior in a reactor resemble those of solutions and because a well-agitated suspension or slurry is virtually homogeneous. Fuels for these reactors thus range from pure Newtonian liquids to paste fuels.

As in other sections, subcritical assemblies, exponential experiments, test loops, and similar concepts will not be treated in a separate chapter because they do not strictly fit the definition of reactors. They will, however, be discussed wherever pertinent. The various critical experiments that have been part of the development of reactors are examples.

Some reactors that embody concepts that strictly fit the definition of homogeneous aqueous solutions or suspensions are discussed elsewhere because they have some other distinctive feature.

Examples are pulsed and excursion reactors. The Russian pulsed reactor is a high-flux research reactor, with an aqueous solution of enriched uranium salt as fuel. The Solution Type Pulse Reactor (STPR) of Atomics International (AI) has highly enriched uranyl sulfate solution for fuel. AI's Kinetic Experiment Water Boiler (KEWB) is, as the name designates, a "water boiler," but it is also a pulsed reactor that can sustain large power excursions. All of these reactors will be described in Part III of the catalog.

The concepts in this section are discussed in the three remaining chapters:

Chapter 2 Research and Testing Reactors
Chapter 3 Non-Boiling Power and Breeder Reactors
Chapter 4 Boiling Power and Breeder Reactors

Most of the research and testing reactors in Chapter 2 are of the misnamed "Water Boiler" type--the water does not boil. They were so named because of the boiling appearance caused by radiolytic gas bubbles. Some others, however, are included. Chapters 3 and 4 cover reactors designed to yield power and to breed fuel. They differ in whether or not heat is removed by boiling in the core.

Obviously, divisions between chapters are not always sharp--as between Chapters 2 and 3. To a large extent the divisions are according to the purpose of the reactor.

Table I[3] lists the most significant homogeneous aqueous reactors with their characteristics and applications. This section generally follows the pattern used in the table.

Early History

Aqueous solutions of uranium salts were used to investigate atomic fission not long after the discovery, by Hahn and Strassmann in 1938, that neutron bombardment could break up the uranium nucleus, and after the

TABLE I Homogeneous Reactor Types and Applications

Reactor designation	Power level range, Mw heat	Fuel solution or suspension	Application
Water boiler	0-0.05	Enriched UO_2SO_4 or $UO_2(NO_3)_2$ in H_2O	University nuclear research and training
Homogeneous research reactors	800-2000	Enriched UO_2SO_4 in D_2O	Nuclear research at ultra-high thermal-neutron fluxes
U^{235} burners	40-500	Enriched UO_2SO_4 in H_2O or D_2O	Small- to large-scale power plants in high-fuel-cost locations; mobile power plants
LAPRE type power reactors	1-100	Enriched UO_3 dissolved in 60 wt.% phosphoric acid Enriched UO_2 dissolved in 95 wt.% phosphoric acid	Remotely located small- and intermediate-scale power plants
One-region power converters	500-1000	Slightly enriched UO_3 in D_2O	Large-scale power production
One-region Pu producer	1000-2000	Slightly enriched UO_2SO_4 in D_2O [with or without added $Li_2^7(SO_4)$]	Dual-purpose power plus plutonium production
Two-region Pu producer	500-1500	Enriched UO_2SO_4 in D_2O (core) Depleted UO_2SO_4 in D_2O (blanket)	Dual-purpose power plus plutonium production
One-region thorium breeder	500-1500	Enriched U^{235} or U^{233} oxide plus ThO_2 in D_2O	Large-scale power production
Two-region thorium breeder, solution core	200-1000	Enriched U^{235} or U^{233} as UO_2SO_4 in D_2O (core) plus ThO_2 in D_2O (blanket)	Large-scale power production and U^{233} breeding or U^{235} to U^{233} conversion
Two-region thorium breeder, slurry core	200-1000	Enriched U^{235} or U^{233} oxide plus ThO_2 in D_2O (core) plus ThO_2 in D_2O (blanket)	Large-scale power production and U^{233} breeding or U^{235} to U^{233} conversion

interpretation of the significance of their work by Meitner and Frisch.[4] Von Halban, Joliot, and Kowarski placed a uranium solution in a large vessel, in the center of which was a source of neutrons. Around it they placed devices to detect neutrons and count them as they were formed. When they substituted for the uranium solution another that was very similar but contained no uranium, they found that many more neutrons were produced when the neutron source was placed in the solution of uranium.[5,6]

This experiment demonstrated that neutron multiplication would take place in an aqueous solution of uranium salt; it was among the first subcritical or exponential reactors. Neutrons from a neutron source were multiplied, but there was not a self-sustaining or chain reaction because the neutron source was needed to maintain the reaction as well as to start it.

Von Halban and Kowarski continued their experiments in England using a homogeneous suspension of uranium oxide in a vessel immersed in about a ton of heavy mineral oil to serve as a reflector. Although this system was not large enough to be itself capable of maintaining a chain reaction, measurements made enabled the experimenters to show that a self-sustaining system would be possible with between 3 and 6 tons of heavy water; they also showed that, with enriched uranium, a self-sustaining reaction would be possible in light water.

In this experiment, a suspension of uranium oxide and heavy water was contained in a spinning sphere that kept the powder well mixed with the fluid. It provided clear evidence that in a sufficiently large system a chain reaction would be possible.[7] Because the only feasible homogeneous reactor using natural uranium would be one moderated with heavy water, and sufficient amounts of D_2O did not become available until 1943 for use in reactors, early interest in homogeneous aqueous reactor systems was purely academic. Even after the atomic energy program was well under way, work was concentrated on heterogeneous reactors.[1] An important advantage of heterogeneous reactors is that fuel can be clad to prevent corrosion, attrition, and escape of fission products.

Interest in homogeneous reactors was revived early in 1943, when a supply of heavy water in U. S. and Canada was expected. However, slurry reactors took precedence over the solution type because enriched uranium was not then available in sufficient quantity. Furthermore, the only known soluble salts of uranium of sufficiently low cross section to enable the design of a reactor of feasible size and heavy water requirement were uranyl fluoride and uranium hexafluoride. These compounds were considered but rejected, principally because of hydrolysis of the UF_6, corrosion, meager solubility data, precipitation, and instability under radiation. A second reason for the choice of a slurry reactor was the evidence that decomposition of the heavy water would be more severe in a solution than in a slurry. In the solution, fission fragments would be formed in intimate contact with the water. They would, however, be formed inside a solid particle in a slurry.[1] Finally, a slurry would have a higher fuel loading than a solution. In spite of the emphasis on heterogeneous and slurry reactors, however, many concepts for homogeneous aqueous reactors were considered for research, power, and breeding.

References

1. J. A. Lane, H. G. MacPherson, and Frank Maslan, Fluid Fuel Reactors, Addison-Wesley Publ. Co., Reading, Mass., 1958.
2. J. A. Lane, "Aqueous Fuel Reactors," Chapter 19 in Reactor Handbook, Vol. IV, Engineering, 2nd Ed., Stuart McLain and J. H. Martens, eds., Interscience Division, John Wiley and Sons, New York, 1964.
3. Ref. 1, p. 12.
4. Gordon Dean, Report on the Atom, 2nd Ed., Knopf, New York, 1957, pp. 242-83.
5. H. Von Halban, Fredric Joliot, and L. Kowarski, "Liberation of Neutrons in the Nuclear Explosion of Uranium," Nature, 143, No. 3620, pp. 470-71, March 18, 1939.
6. L. Bertin, Atom Harvest, Secker and Warburg, London, 1955, pp. 42-44.
7. Ref. 6, p. 57.

Chapter 2 Research and Testing Reactors

The reactors taken up in this chapter are characterized by their purpose--to give experimental data, primarily through the nuclear radiation they produce. They are thus distinguished from those experimental reactors that are used primarily to obtain data for the design of a power reactor. Such experimental reactors will be described in connection with the pertinent power reactors. The USAEC Division of Technical Information has defined a research reactor as "any reactor whose nuclear radiations are used primarily as a research tool for basic or applied research regardless of operating power level. May include facilities for testing reactor materials."[1] Most of these research reactors are of the "water-boiler" type, but there are several others.

Zero-Power Reactors

A few reactors of extremely low power or "zero power" deserve to be treated separately, rather than with power reactors as the subcritical assemblies and critical experiments are discussed. They are reactors useful in their own right rather than as part of the development of another reactor. Zero-power reactors have been defined by the National Research Council as experimental nuclear reactors operated at low neutron flux and at a power level so low that not only is no forced cooling required but also fission product activity in the fuel is sufficiently low to allow the fuel to be handled after use without serious hazard.[2]

NASA Zero-Power Reactor

This reactor was planned for performing critical experiments, measuring reactivity effects, serving as a neutron source, and being a training tool. Varying core configurations of solutions of highly enriched uranyl fluoride in light water would operate at a maximum power of 10 watts.[3] Two versions, NASA ZPR-I and NASA ZPR-II have been listed, with the second described as an enlarged version of the first.[4] The power levels, however, apparently are the same. In December 1963, both were being used for critical experiments, with 1959 being given as a startup date for ZPR-I and 1963 for ZPR-II.[1] ZPR-I was described as "critical experiments for the NASA Test Reactor," ZPR-II as "NASA Zero-Power Reactor II, solution type critical." Plans were for operating both in the same building, but with only one reactor operating at a time.

Some of the data obtained from this reactor were used in a computer program to compute the age of fission neutrons in water.[5]

The design of this reactor utilized earlier work at Oak Ridge on the criticality of aqueous solutions of uranium-235 salts.[6,7]

Highly Enriched Homogeneous Reactor

This reactor,[8] described by Brown, et al, of Westinghouse APD, is of lower power than that of NASA. It contains an extremely dilute aqueous solution of uranyl nitrate, in which the enrichment is greater than 90 percent. It is contained in a stainless-steel tank, 72 inches high and 36 inches in diameter.

Proserpine

This French "reactor" is actually a critical facility, but is in the form of a small water boiler and thus is discussed here. It has been described in several references.[9,10] It is designed to study criticality problems of plutonium solutions, and it utilizes a fissionable solution of plutonium sulfate, $Pu(SO_4)_2$, in 0.5N H_2SO_4 in light water. The reflector is beryllium oxide and graphite.

Water Boilers

A "water boiler" is a small reactor with a core consisting of a container of fuel solution. It is equipped with controls, reflectors, heat-transfer system, and other auxiliaries as required. The fuel is a soluble salt of uranium or plutonium; the moderator-solvent is light or heavy water.

The core is a stainless-steel sphere, about one foot in diameter, containing the fuel solution.[11] Because a natural uranium light water solution cannot be made critical, the uranium is enriched in U^{235}. The fuel solution is cooled by water circulating through coils inside the core. Normally this cooling keeps the solution temperature below about 80°C and, because the operating pressure is near atmospheric, no actual boiling occurs. One water boiler, SUPO, was, however purposely operated under boiling conditions in a stability study.[12]

A fuel-handling system adds fuel solution to the core and recovers solution expelled by any operation above design power. Hydrogen and oxygen evolved in operating at relatively high power are catalytically recombined and the water is returned to the core. The core is surrounded by a graphite or beryllium oxide neutron reflector with a concrete shield around the reflector. Control is by neutron-absorber rods, usually containing boron or cadmium, with passages or thimbles for the rods in the core or reflector. Usually the reactors have experimental facilities, such as thermal columns and beam tubes.[11]

The water boiler has evolved through many stages,[13-18] and various types are now marketed by several manufacturers.[19,20]

The LOPO Series

This water boiler, or LOPO-HYPO-SUPO concept, is the prototype for all water-boiler reactors. It was important in World War II in that it provided vital data for the atomic bomb project.

In all these reactors, power and breeding are not primary objectives. LOPO originally was built "to gain experience in the operation and control of a chain-reacting assembly while using a minimum of active material... Its main job was the mass production of neutrons..."[21]

The "water boiler" at Los Alamos was planned in 1943 to test certain theoretical calculations[13] when an enriched uranium supply seemed possible. R. F. Christy was responsible for the critical-design calculations for this reactor.[14,15,22] Christy's reactor design consisted of a noncorrosive stainless-steel shell, one foot in diameter, containing enriched uranium or plutonium as a water solution of a suitable salt, such as uranyl sulfate or nitrate, with a neutron reflector of beryllium oxide bricks occupying a cube of about four feet on an edge. The fuel solution circulated by convection, and about 10 kw(t) was extracted by a coiled tube carrying cold water through the inside of the reactor. Control was exercised through a vertically moving control rod.

Preliminary plans and calculations for such an operation were completed by September 1943, and a critical experiment, LOPO-CX, was carried out in 1943-44. Ten kilowatts was arbitrarily chosen as the operating level. It appeared, however, more advisable to construct first a reactor of much lower power, known as the Low Power Water Boiler (LOPO), because it eliminated heavy shielding requirements and minimized the problems of uranium compounds going out of solution, gas evolving from decomposition of the water by fission fragments, and fission fragments contaminating the solution. The plans for LOPO were completed in November 1943, assembly proceeded through the spring of 1944, and the assembly went critical in 1944.[13,21]

LOPO was one of the first aqueous-solution reactors using enriched fuel. It used 14.5 percent enriched uranium (565 grams of uranium-235 as uranyl sulfate) dissolved in ordinary water. The solution was contained in a Type-347 stainless-steel sphere one foot in diameter, with a beryllium oxide reflector and a cadmium control rod. The lack of a shield and cooling system limited the heat power level of LOPO to 50 milliwatts.

LOPO was later modified to become the High Power Water Boiler (HYPO), which went critical in December 1944. It operated at a peak power of about 6 kw. Major modifications included the addition of a shield and an internal cooling system consisting of a single coil, which carried cooling water through the core. Additional experimental holes were added, the uranyl sulfate fuel solution was replaced by uranyl nitrate because an extraction method to remove fission products was known only for the nitrate solution at that time, and the control system was made more elaborate.[21,23,24]

Because higher neutron fluxes and more research facilities than were available from HYPO were desired, the reactor was further modified starting in April 1949. Completed in March 1951, it became SUPO, the Super Power Water Boiler. Major changes included an increase in operating power level from 5.5 kw to a minimum of 45 kw by replacing the cooling coil by three stainless-steel tubes, replacing the beryllium portion of the reflector by graphite, changing the reactor solution from 15 percent enriched uranyl nitrate to one of 88.7 percent enrichment, and adding new experimental holes and vertical control rods.[23,25,26,27]

Atomics International Reactors

Atomics International's 5-watt Water Boiler Neutron Source (WBNS) was built and first operated at Downey, California; it went critical in 1952. It was later converted to a 2.5-kw reactor, AE-6. The AE-6 was used as a source of thermal neutrons to study nuclear behavior of heterogeneous subcritical assemblies and the irradiate small samples in neutron fluxes up to 10^{11} neutrons/cm^2/sec.[28-33]

The fuel solution was contained in a one-foot diameter stainless-steel sphere, initially with no cooling system (as in LOPO) because of the very low power level, 1 to 2 watts, but cooling coils were provided in case of higher-power operation. They were used in the later 2-kw operation. In the loading sequence of the special fuel system, the total amount of distilled water was added to the core tank, then the first aliquot of fuel (one-half of critical mass). Next, gas was bubbled through the solution to mix the fuel and distilled water, and a part of the resulting solution was raised into a mixing tank. The next measured aliquot of fuel was added and the process repeated. The fission gases were disposed of by a special gas-disposal system with accumulator, gages, and valving. A graphite reflector two feet thick, control rods, and a concrete shield were used.

Other Atomics International reactors similar to AE-6 include the Livermore Water Boiler Neutron Source,[34,35] starting in 1953; the University of California at Los Angeles (UCLA) Medical Reactor and the Armour Research Reactor, which went critical in 1956; the Danish Reactor-1 and the Japan Research Reactor of 1958, the Italian CESNEF Reactor of 1959; the Walter Reed Army Medical Center Homogeneous Reactor, Washington D.C., which went critical in 1961; and the Los Alamos Medical Reactor, which was scheduled for construction starting in 1958 but apparently never went critical.[16,36,37]

The Armour Research Reactor[16,32] was the first operating industrial reactor in the U.S. It was operated by Illinois Institute of Technology in Chicago and financed by 24 industrial users. It went critical on June 24, 1956. The reactor operated until 1958 at 10 kw power, an interim power limit "pending the results of kinetic experiments ... carried out on an identical machine by Atomics International for the AEC."[38] The design for the NAA Medical Reactor[39] was similar to that of the Armour Research Reactor, as were those of the AI Models L-54 and L-55.[16]

Other reactors resulted from another Atomics International water boiler type, the "Laboratory Reactor," beginning with the L-47 prototype,[16] a predecessor to the L-77. Reactors of this type are the University of Wyoming Research Reactor and the Puerto Rico Reactor at the Puerto Rico Nuclear Center, both of which went critical in 1959.

These L-77-type reactors are stainless-steel spherical vessels containing a critical mass of 20 percent enriched uranyl sulfate fuel and are controlled by cadmium-lead rods working through vertical thimble tubes; lead is used as a reflector.[16,32,38,40] Flora comments that the "Los Alamos 'Water Boiler' was chosen as a starting point since this type of research reactor has more years of successful operating history to its credit than any other type of research reactors."[41]

The unit was so designed that the factory "package" could be shipped and assembled without difficulty and could be used in existing facilities and without special staff. One L-77, known as the DEMAG Reactor, was operated at the Second Geneva Conference in 1958 as part of the commercial exhibit. Another L-77 was included in the U. S. Atomic Energy Commission's exhibit at Beirut, Lebanon, in its Atoms for Peace Program.[42]

Various modifications of this reactor were made, such as the 5-Watt Laboratory Reactor[41,43] or the Small 5-Watt Laboratory Reactor.[41]

Other Water Boilers

Water-boiler research reactors, in addition to those of Atomics International, include those designed at universities and miscellaneous types. North Carolina State College at Raleigh was the first university to design (R. L. Murray), build, and operate a reactor.[44] A water boiler was chosen because details of this type were the first to be released by the AEC. The North Carolina State College Reactor (NCSR or NCSCR), also known as the Raleigh Research Reactor (RRR-1), went critical in 1953.[32,45] NCSCR-1,[32] NCSCR-2,[32] and NCSCR-4[46] were all the water boiler type; the NCSCR-3, however, was a 10-kw pool type completed in March 1960. NCSCR-4 still is in use at North Carolina State, and a similar type is in use at the University of Wyoming.[47,48] The NCSCR reactors are substantially the same as the original LOPO-HYPO-SUPO, except that a cylindrical core is used instead of a sphere. The Utah Water Boiler reactor[32] is a reactor designed at the University of Utah and described by Borst and Mong.[49,50] It is intended for a university wishing to operate with the utmost simplicity and safety. More information on university reactors is given in the records of the University Reactor Conferences.[47]

Two French homogeneous aqueous reactors for special-purpose testing have recently been reported.[51] Few details are given. Alecto I and Alecto II, at Saclay, France, operate at a few watts, with a maximum thermal flux of 10^8 neutrons/cm^2/sec. Alecto I, which became critical in 1961, uses uranium nitrate solution in light water. Alecto II, which became critical in 1962-63, uses a solution of plutonium nitrate, also in light water. Both were designed, built, owned, and operated by CEA (Centre d'Etudes Nucléaires de Saclay).

Miscellaneous water boilers on which there is little information are the Chinese Homogeneous Research Reactor at Tientsin,[52,53] and that of the Gamma Corp., Mansfield, Massachusetts.[54]

A modified form of the water boiler is the Test Tube Reactor developed at Los Alamos.[55] This reactor, so-called because of its cylindrical shape, is related to the LAPRE-1 (Los Alamos Power Reactor Experiment Number 1) which has also been called a "test-tube reactor." The fuel solution, highly enriched uranyl sulfate or phosphate in light water, moves by convection up a hot central tube and returns through the surrounding annulus, which contains coolant coils. This reactor has a power rating of 100 kw.

Babcock and Wilcox also designed a similar modified water boiler for nuclear and engineering research.[56] The design power was 50 kw, with enough heat-transfer surface in the core to make higher power possible. The solution--20 liters of uranyl sulfate in light water--is in a cylindrical container with a hemispherical bottom.

Irradiation and Testing Reactors

In addition to water boilers, other aqueous solution reactors have been designed for diverse irradiation and testing purposes. Some are operated at higher power than water boilers, and two are not strictly homogeneous, in that they contain solid moderators.

Several aqueous-solution reactors were conceived as neutron producers. An exploratory study of reactors for gamma irradiation in food sterilization[57] indicated that homogeneous reactors have no advantage over other types as irradiation sources for food sterilization if they are used only for this purpose. If, however, they can be used both for irradiation and power production, they have several economic advantages. A meat-irradiation facility has been tentatively suggested in another report.[58] This facility would be a 15-Mw reactor with a solution of uranyl sulfate and copper sulfate in light or heavy water. It resembles the Homogeneous Reactor Test. (See Chapter 3.)

The Homogeneous Research Reactor, also known as the Aqueous Homogeneous Research Reactor, is a single-region reactor with 8 percent enriched uranyl sulfate solution in heavy water as fuel-moderator-coolant. It was intended to operate at 500 Mw(t) or 125 Mw(e) and to produce a thermal flux of 5×10^{15} neutrons/cm^2/sec.[59,60] It was concluded that the design could proceed on the basis of the assumed parameters. Successful operation was expected, with no technological breakthroughs required but with many attendant problems.[61]

A later study at Aeronutronic Systems, Inc., considered both heterogeneous and homogeneous reactors for a flux of 2×10^{15} neutrons/cm^2/sec. A homogeneous Advanced Engineering Test Reactor (AETR) based very largely on the design of the HRE, HRT, and the Homogeneous Research Reactor was selected as preferable.[62] Removal of fission products and addition of fuel would be continuous. To maintain the total contained excess reactivity at a constant level, the solution, with 10 percent uranium-235, would be contained in a stainless-steel sphere 8 feet in diameter.

A reactor suggested for high-flux irradiation in chemical processing employs a subcritical zone for exterior circulation of fuel.[63] A homogeneous aqueous solution of uranium phosphate is one form of fuel specified. The fuel flows from the reactor to a cooling zone, and then back to the reactor. The passages into and out of the reactor contain neutron-absorbing material, like boron, to keep the material subcritical outside of the reaction zone. This arrangement permits very high fluxes--above 10^{13} neutrons/cm^2/sec--and high heat generation--more than 20 Mw per pound of fissile material. Several variants are described, with such other fuels as gases, liquid metals, molten salts, or fluidized solids.

A high-flux homogeneous reactor described by Hibshman utilizes enriched uranyl sulfate in light water to give a power of 28 Mw.[64] The fuel is fed to the core and withdrawn through many inlet and outlet tubes.

Two reactors with aqueous solutions and solid moderators have been suggested.

The ASTRA (Advanced Scientific Techniques Research Associates) Advanced Engineering Test Reactor also used ideas from the Oak Ridge Homogeneous Reactor Project. It, however, has a structural core. This reactor contains highly enriched uranyl sulfate in heavy water solution, and a moderator-reflector of graphite blocks.[65]

The High Flux Research Reactor for Large-Volume Irradiations is an engineering test reactor with a fuel-coolant solution of uranyl sulfate in light water. The solution is circulated through the graphite moderator in parallel tubes. The reactor is designed to produce 120 Mw.[66]

<u>Evaluation and Status</u>

As the water boiler is the most common research reactor, its future possibilities have received the most attention. Breazeale[67] has compared water-boiler and the swimming-pool reactors--the two designs suitable for a low-power research reactor. The water boiler, using a light-water solution of uranyl sulfate and surrounded on all sides by a good reflector like carbon or beryllium, will have a critical mass of less than 1 kg of uranium-235. The swimming-pool type with 1-kg critical mass would be inflexible and, because of its small heat-transfer surface, could be operated only at low power. But at 2-kg critical mass, the swimming-pool type has greater power capability and flexibility than the water boiler of 1 kg. Presumably, 2 kg enriched uranium is available for this extra flexibility; moreover, it is easier to use a water shield with the swimming-pool reactor than with the water boiler, which needs a shielded tube to carry off the radioactive gases.

Lane[68] has pointed out that water boilers have limited use as research reactors, although a few years ago they seemed to be promising competitors to the swimming-pool and tank-type reactors. Some of the difficulties are:

a. The use of water as a fuel-bearing medium limits operating temperature and neutron flux.

b. Severe corrosion, both from circulating solutions and radiation, can damage equipment, resulting in such conditions as loss of radioactive solution from a reactor tank. Such leakage occurred in the Raleigh Research Reactor.

c. Unsafe conditions may result from excessive reactivity changes.

d. Hazards arise from the explosive hydrogen-oxygen mixture produced by radiolytic decomposition of water.

e. Highly trained technical workers, especially chemists, are required.

Water boilers and similar solution reactors have many of the advantages and disadvantages discussed by Lane in a recent work on aqueous reactors.[69]

Solution reactors have generally fallen out of favor for research uses, with few new ones being built. However, the water-boiler reactor at Walter Reed Hospital went critical as late as 1961 and was in operation by the fall of 1962.[70,71] Also, in early 1963, the AEC announced that it planned to issue a permit to the University of Nevada for construction of an Atomics International Model L-77 reactor.[72]

water-boiler and the swimming-pool reactors—the two designs suitable for a low-power research reactor. The water boiler, using a light-water solution of uranyl sulfate and surrounded on all sides by a good reflector like carbon or beryllium, will have a critical mass of less than 1 kg of uranium; i.e. a swimming-pool type with less critical mass would be infeasible and, because of its small heat-transfer surface, could be operated only at low power. But at 2-kg critical mass, the swimming-pool type has a greater power capability and flexibility than the water-boiler of comparable. Presumably, 2 kg enriched uranium is available for this purpose. Moreover, it is easier to use a water shield with the swimming-pool reactor than with the water boiler, which needs a shielded sub to carry off the gaseous radioactive products.

Lane[6] has pointed out that water boilers have limited use as research reactors, although a few years ago they seemed to be promising competitors to the swimming-pool and tank-type reactors. Some of the difficulties are:

a. The use of water as a fuel-bearing medium limits operation temperature and neutron flux.

b. Severe corrosion, both from circulating solutions and attacking with decay equipment, resulting in such conditions as leakage of radioactive solution from a reactor tank. Such leakage occurred at the Raleigh Research Reactor.

c. Unstable conditions may result from excessive reactivity changes.

d. Hazards arise from the explosive hydrogen-oxygen mixture produced by radiolytic decomposition of water.

e. Highly trained technical workers, especially chemists, are required.

Water boilers and similar solution reactors have many of the advantages and disadvantages discussed by Lane in a recent work on aqueous reactors.

Solution reactors have generally fallen out of favor for research use with few ones being built. However, the water-boiler reactor of Walter Reed Hospital went critical as late as 1961 and was in operation by the fall of 1962. Also, in early 1963 the AEC announced that it planned to issue a permit to the University of Nevada for construction of an Atomics International Model L-77 reactor.

D A T A   S H E E T S

RESEARCH AND TESTING REACTORS

No. 1 Zero-Power Reactor

(ZPR-I and ZPR-II)

NASA Lewis Research Center

References: NACA-RM-SE-57-F-28; TID-8200 (9th rev.); AEC Press Release, Mar. 23, 1962.

Originators: Laboratory staff.

Status: December 1963; critical experiments being carried out.

Details: Two reactors, ZPR-II stated to be enlarged version of ZPR-I, but power is given as the same; otherwise also apparently the same. Thermal neutrons, steady state, burner. Fuel-moderator-coolant: solution of highly enriched uranyl fluoride in H_2O. Concentration will change according to experiment. Reflector will also vary. Solution added as needed through line from solution room. Core vessel: cylinder, open at the top. Shielding: 54-in. thick concrete. Control: amount of solution in reactor and inherent negative temperature coefficient of the solution. Control rod used for safety only. Emergency dumping into safe-geometry storage tanks provided as additional safety measure. Power: normally 1/10 watt, with maximum of 10 watts; neutron flux up to $10^8/cm^2/sec$.

Code:	0313	13	31201	44	624	74	83779	921	101
							84677		
							811XX		

No. 2 Highly Enriched Homogeneous Reactor

WAPD

Reference: WAPD-128.

Originators: J. R. Brown, B. H. Noordhoff, and W. O. Bateson.

Status: Critical experiments performed, May 1955.

Details: Thermal neutrons, subcritical reactor. Fuel-moderator-coolant: solution of uranyl nitrate in H_2O; more than 90% U^{235}; ratio of H to U^{235}, approximately 1700. Reflector: Lucite plastic at bottom of tank, which is a stainless-steel cylinder 36 in. diameter, 72 in. high, with 1/8 in. thick walls. Reactor tank inside a concentric cylinder, 52 in. diameter, 70 in. high, filled with water. Control: two safety and one regulating rod. All move vertically. Each is hollow Cd cylinder between concentric tubes of stainless steel. Emergency dumping provided as additional safety measure.

Code:	033X	13	31201	44	624	711	81112	921	101
							84677		
							83779		

No. 3 Proserpine Critical Experiment

Commissariat á l'Énergie Atomique, France

Reference: Proc. 2nd U.N. Int. Conf., 12, pp. 539-562.

Originator: J. Bertrand et al.

Status: Critical in 1958.

Details: Thermal neutrons, critical experiment. Fuel-moderator-coolant: concentrated solution of plutonium sulfate in H_2O. Reactor is reflected, single-region. Reflectors: 27.5-cm thick layer of BeO next to the core tank and 50-cm thick layer of graphite next to the BeO. Core vessel: cylindrical stainless-steel tank, 25 cm in diameter and 30 cm high, with a wall thickness of 1 mm. Outside the core tank is one of aluminum designed to retain solution leaks. Control: parallel horizontal regulating rods. One, for fine regulations, is a stainless-steel cylinder. The other, for compensation, is a stainless-steel cylinder lined with Cd. Rods are driven by a mechanism outside the pile. Two horizontal safety rods, externally tangent to the tank, are of an alloy of Al and Mg. They are lined with Cd and filled with boron carbide beads.

Code: 0313 13 31201 46 624 711 81212 921 101

No. 4 LOPO, Low Power Water Boiler

LASL

References: AECD-3063 (LADC-819); AECD-3059, (Rev. Sci. Instruments, 22, No. 7, pp. 489-499, July 1951); Proc. 1st U.N. Int. Conf., 2, pp. 372-391, Lane et al, Fluid Fuel Reactors, pp. 341-346.

Originators: D. W. Kerst. Calculations of critical mass first performed at Los Alamos by R. F. Christy.

Status: Startup, May 1944. Replaced by HYPO November 1944.

Details: First "water boiler"--so-called because of gas bubbles formed by radiolytic decomposition of water--and first reactor to use enriched fuel. Thermal neutrons, steady state, burner, some conversion. Fuel-moderator-coolant: solution of uranyl sulfate (14.5% U^{235}) in H_2O. Solution contained in a thin-walled stainless-steel sphere enclosed in thermostatted housing at 39 $\pm$ 0.01°C. Control: Cd-wrapped brass cylinder, with another brass tube fitting snugly over the Cd and moving vertically in the reflector. Inherent safety feature: negative temperature coefficient. Emergency dumping of solution provided as additional safety measure. Over-all reactor size, 3 ft square by 4 ft high. Power: 50 milliwatts.

Code: 0311 13 31201 43 624 744 81152 921 101
84677

No. 5 HYPO, High Power Water Boiler

LASL

References: AECD-3065 (LA-394); AECD-3059 (Rev. Sci. Instruments, 22, No. 7, July 1961, pp. 489-499); Lane et al, Fluid Fuel Reactors, pp. 341-6.

Originators: L. D. P. King, advised by E. Fermi on basis of LOPO.

Status: Critical, December 1944; Alterations began April 1949 to replace HYPO with SUPO. Dismantled, 1950.

Details: Same as LOPO except: (1) 14% enriched U^{235} as uranyl nitrate instead of sulfate; (2) natural convection plus internal cooling coil with circulating water; (3) thicker sphere with horizontal 1-in. pipe or "glory hole" for access to highest possible neutron flux; (4) two additional Cd control rods; (5) addition of graphite thermal column; (6) inclusion of γ-ray and neutron shields; and (7) elimination of hydrostatic control system and storage reservoir for solution. Solution temperature: 85°C. Peak power of 6 kw.

Code:	0311	13	31201	43	624	744	81152	921	105
							84677		

No. 6 SUPO, Super Power Water Boiler

LASL

References: LA-1301; AECD-3287 (LADC-1081); Lane et al, Fluid Fuel Reactors, pp. 341-6.

Originators: L. D. P. King, group director.

Status: In operation, June 30, 1961. Modifications (of HYPO) first phase completed February 1950. Modifications second phase completed March 1951.

Details: Modified HYPO First Phase: Fuel enrichment increased to 88.7% U^{235} (as uranyl nitrate). Be reflector replaced by graphite. Interior of sphere rebuilt to include 3 stainless-steel coolant coils. Coolant temperature: 130°F; atmospheric pressure. Second thermal column added. Two more control rods of sintered B^{10} in thimbles added, 3 Cd sheets tangent to sphere included. Reactor size (outside of shield): 15 ft x 15 ft x 11 ft high. Power increased to 30 kw. Reactor was run for 10,000 kw-hr before second phase modifications were begun. Power then increased to 45 kw. Other alterations include rebuilding of original thermal column, construction of recombination system to handle offgases, and construction of shielded solution-handling system. Estimates also given for intermediate and fast neutrons.

Code:	0313	13	31201	44	624	711	81111	921	105
							81112		

No. 7 Supo Model II

LASL

References: U.S. Patent 2,961,391; Proc. 1st U.N. Int. Conf., 2, 1955, pp. 372-91.

Originator: L. D. P. King.

Status: Patent granted, November 1960.

Details: Same as SUPO with following changes: core shape changed to 66-in.-high, 12-in.-diameter cylinder; single control rod (presumably Cd); gas-circulation rate in recombination system increased to 200 ml/min; sizes of catalyst chamber and heat exchanger increased to produce higher power level of 400 kw. Lower part of the cylinder is critical region, filled with solution to a height equal to its diameter (12 in.). Fuel is preferably uranyl nitrate, but sulfate may be used.

Code: 0313 13 31201 44 624 711 81112 921 105
84677

No. 8 WBNS, Water Boiler Neutron Source

Atomics International, A Division of North American Aviation, Inc.

References: Science, 119, pp. 9-15, Jan. 1, 1954; NAA-SR-839; AECU-2900, pp. 183-191.

Originators: R. Chalker and M. E. Remley.

Status: Put into operation April 1952. Dismantled July 1956.

Details: Based on data from LOPO and HYPO. Thermal neutrons, steady state, burner. Fuel-moderator-coolant: solution of 90% enriched U^{235} as uranyl nitrate in distilled H_2O. Cooling coils provided (but not used because of low power) in 1-ft-diameter stainless-steel spherical core tank. Reflector: cylindrical, of graphite blocks adjacent to sphere; also serves as thermal column. Design pressure: 300 psi. Control: 2 safety rods--strips of boral attached to Al channel to form I-beams; gravity-actuated with horizontal movement, and 1 fine (pipe) and 1 coarse (also I-beam) Cd control rods. Reactor housed in tank 6 ft high, 5 ft in diameter. Operating power: 1 watt.

Code: 0313 13 31201 44 624 711 81211 921 101
81212

No. 9 AE-6 Water Boiler Neutron Source (Model L-6)

Atomics International, A Division of North American Aviation, Inc.

References: TID-2503, pp. 69-76; IAEA Directory of Nucl. Reactors, 2, 1959, pp. 181-6.

Originators: G. L. Blackshaw and C. H. Skeen.

Status: Critical, November 1956; in operation, June 30, 1961.

Details: Basically a conversion of WBNS to power of 2.5 kw. Changes include 93.11% enriched U^{235} as uranyl sulfate instead of nitrate; cooling system employed with distilled water (inlet temperature, 42°F; outlet, 52°F). Two safety rods--strips of boral attached to an Al I-beam, 1 shim rod of Cd attached to Al I-beam, 1 regulating rod of Cd-filled stainless-steel tube. Horizontal movement: tangential to core. Gas-handling system and heavier permanent shielding around core and reflector added. (AE-6 [or L-6] was the prototype for L-55 operating as Danish Research Reactor No. 1.)

Code: 0313 13 31201 44 624 711 81211 921 105
81212
84677

No. 10 Livermore Water Boiler Neutron Source, LIWB or North American Aviation Research Reactor Model L-3

(also Cal. Research & Development CR&D Reactor; Livermore Research Reactor LRL)

Livermore Research Laboratory

References: NAA-SR-MEMO-784; NAA-AER-1023 (extended version*).

Originator: NAA staff.

Status: Startup, November 5, 1953. In operation, June 30, 1961.

Details: Higher-power version of WBNS. Same except for fuel, uranyl sulfate instead of nitrate. Coolant: distilled water. Sphere: 12-1/2-in. diameter. Two safety rods and 1 coarse control rod of flattened stainless-steel tubing packed with boron carbide. One fine control rod of boron carbide. Power: 500 watts. Unlike other models of WBNS, LIWB has completely closed-cycle sweep gas and recombination system.

*Extended and revised version produces more power (50 kwt), has larger thermal column, and uses bismuth instead of lead for the gamma shield.

Code: 0313 13 31201 44 624 711 81211 921 105
81221

No. 11 Armour Research Reactor, (ARR)

(Atomics International Model L-8)

Armour Research Foundation*

References: Proc. 2nd U.N. Int. Conf., 10, pp. 393-403; Proc. 2nd U.N. Int. Conf., 10, pp. 404-418; IAEA Directory of Nuclear Reactors, 2, pp. 165-70.

Originators: Staff of Atomics International, a Division of North American Aviation, Inc.

Status: Critical, June 1956; 10 kw operation, December 1956; 100 kw operation, January 1959. In operation, June 30, 1961.

Details: Thermal neutrons, steady state, burner. Fuel: 88.14% enriched U^{235} as uranyl sulfate in light water. Coolant: distilled light water circulating through cooling coils inside core. Working pressure below atmospheric. Reflector: graphite blocks. Stainless-steel spherical core vessel 12.4 in. diameter. Four control rods of boron carbide clad with stainless steel, 3 electrically driven, (rack & pinion type) 1 servo-system operated, moving vertically. Because of urban location (Chicago), system of barriers (shields) arranged: primary, closed core vessel and gas-handling system of stainless steel; secondary, core vessel and steel lining of sub-pile room, gas-tight doors, in Al cylinder; tertiary, reactor room, air-lock, gas-tight doors, sealed conduits and water-floodable ducts for isolation of room ventilation system. Can run at 50 to 100 kw. ARR is the prototype for Atomics International Model L-54. (First industrial research reactor). Vertical and horizontal experimental facilities.

Code: 0313 13 31201 44 624 711 81111 921 105

*Now Illinois Institute of Technology Research Foundation.

No. 12 Medical Reactor

NAA

Reference: NAA-AER-1180.

Originator: NAA staff.

Status: Conceptual design, 1955; design used for the LASL Medical Reactor; similar to L-8 (ARR) design.

Details: Thermal, steady state, burner. Fuel-moderator solution is enriched UO_2SO_4 in light water, cooled by chilled H_2O circulating in cooling coils in the core. Spherical core, 1-ft diameter, is surrounded by a graphite reflector (5 ft x 5 ft x 8 ft long). The core has 5 re-entrant thimbles, 4 for control and safety rods of B_4C pressed into a vertical cylinder and 1 as an exposure facility. A secondary Al enclosure around the stainless steel pressure vessel insures against any leaks. Maximum fuel-solution temperature at full power (50 kw) is 80°C. The design includes extensive exposure facilities for medical research.

Code: 0313 13 31201 44 624 711 81111 921 105

No. 13 Research Reactor Model L-54

Atomics International, A Division of North American Aviation, Inc.

Reference: Proc. 2nd U.N. Int. Conf., 10, pp. 404-18.

Originator: AI staff.

Status: Concept used for five operating reactors: JRR-1, Japan; FRB (BER), Berlin, Germany; FRF, Frankfurt, Germany; CESNEF, Milan, Italy; and Walter Reed Medical Research Reactor.

Details: Revision of ARR (L-8) design, using 20% enriched uranium in uranyl sulfate. Primarily burner, but some internal conversion. Core size increased to 15.75 in. diameter to contain larger amount of fuel solution. Wet-type gas-handling system. Other aspects indentical with ARR (L-8). Operating power: 50 kw.

Code: 0311 13 31201 43 624 744 81111 924 105

No. 14 Research Reactor Model L-55

Atomics International, A Division of North American Aviation, Inc.

Reference: Proc. 2nd U.N. Int. Conf., 10, pp. 404-18.

Originators: J. W. Flora et al.

Status: Concept used for one reactor, the Danish DR-1.

Details: Variation of AE-6 (L-6). Thermal neutrons, steady state, burner. Fuel can be uranyl sulfate (20% to 90% U^{235}) in light water. Coolant: distilled water. Reflector: graphite. Core vessel: stainless-steel sphere, 12.5 in. diameter. Four horizontal control rods (2 safety, 1 shim, 1 regulating) of flattened stainless-steel tubes filled with B_4C powder, tangent to core vessel. Vapor-pressure-type recombiner. Thermal column, 5-1/2 ft square vertical. Power: up to 1.5 kw.

Code:	0313	13	31201	43	624	744	81221	921	105
				44		711			

No. 15 Laboratory Reactor Prototype L-47

Atomics International, A Division of North American Aviation, Inc.

Reference: Proc. 2nd U.N. Int. Conf., 10, 1958, pp. 404-18.

Originator: AI staff.

Status: Startup, August 29, 1957; dismantled, May 1958.

Details: Five-watt prototype unit for L-77. Thermal neutrons, steady state, burner. Fuel-moderator: 89% enriched U^{235} as uranyl sulfate in light water. Cooled by natural convection; no cooling coil. Stainless-steel spherical core vessel 12.5 in. diameter, 12.5 liter capacity. Canopy assembly over core contains recombiner. Reflector: lead shield, 6 in. thick, completely surrounding reactor core and recombiner system. Core reflector assembly supported in middle of shield tank, 8 ft diameter, 8 ft high, filled with water. Sodium octaborate later added to water to suppress water-capture gamma radiation. Control: two vertical Cd control rods. Experimental facilities: 5 through-tubes in the core and 2 beam tubes terminating at the lead surface. Operating power: 1 watt.

Code: 0313 13 31201 44 624 711 81112 921 101

No. 16 L-77 Laboratory Reactor

Atomics International, A Division of North American Aviation

References: Proc. 2nd U.N. Int. Conf., 10, 1958, pp. 404-18; IAEA Directory of Nuclear Reactors, 2, 1959, pp. 205-210.

Originator: Atomics International, Division of NAA.

Status: Startup May 1958; in operation, June 30, 1961.

Details: Same as L-47 except: fuel: 20-90% enriched U^{235}. No cooling coil. Reflector: Pb shot-diphenyl mixture. Core vessel: sphere, 15.75 in. diameter; same canopy assembly for recombiner. Working pressure: 25-28 in. Hg. Containment vessel: stainless steel, 8 ft diameter, 7 ft high. Three-region shield system: inner, pelletized Pb-diphenyl compound; intermediate, paraffin and soluble organic boron compound; outer, borated paraffin and pelletized Pb, enclosing intermediate and inner shields on sides and top. Safety and regulating rods; Cd-sheathed stainless steel sealed in aluminum. Power: 10 watts.

Code:								
0313	13	31201	44	624	711	81112	921	101
			43		744			

No. 17 5-Watt Laboratory Reactor

(Atomics International Aqueous Homogeneous Laboratory Reactor)

Atomics International, A Division of North American Aviation

References: Nuclear Eng., 1, No. 8, November 1956, pp. 344-5; 2nd Nucl. Eng. and Sci. Conf., 2, "Advances in Nucl. Eng."

Originator: AI staff.

Status: Manufactured by AI for educational and small research laboratories.

Details: Smaller and considerably simplified version of ARR, with some details similar to L-47 and L-77. Thermal neutrons, steady state, burner. Fuel-moderator: highly enriched U^{235} as uranyl sulfate in H_2O. Coolant: water, natural conduction and convection; no additional cooling. Core vessel: stainless steel, 12 in. diameter. Hemispherical canopy over sphere houses recombination and overflow chamber. Core and canopy contained in 6 in. thick Pb shield which acts as reflector and primary shield. These are enclosed in Al container, 2 ft diameter and 2 ft 8 in. high. Complete assembly slung from 3 steel tubes in a tank, 8 ft diameter, 8 ft high, of borated H_2O. Control: two Cd cylinders as control and safety rods moving vertically. Power: 5 watts.

Code: 0313 13 31201 44 624 711 81112 921 101

No. 18 North Carolina State College Reactor-1

(NCSCR-1, NCSR-1, RRR-Raleigh Research Reactor)

References: AECU-1986; IAEA Dir. of Nuc. Reactors, 2, 1959, pp. 189-98.
Originator: C. K. Beck.
Status: Critical September 1953; dismantled, June 1955. Modified to NCSCR-2.
Details: Based on HYPO, SUPO, and HRE series. Thermal neutrons, steady state, burner. Fuel: uranyl sulfate with 93% enriched U^{235} in aqueous solution. Coolant: H_2O circulating through 4 stainless-steel, internal cooling coils of hairpin shape. Reflector: 5 ft cube of graphite bars surrounded by Pb. Core vessel: stainless-steel cylinder, 10.75 in. diameter, 9.2 in. high. Control: 2 combined control and safety rods of thin-walled stainless-steel tubes filled with boron powder; 2 shim rods of Cd-filled stainless-steel tubes ("strips") tangent to cylinder. Control rods enter core vertically. Reactor containment vessel: octagon, 17 ft across. Core below ground level. Power: 10 kw (+). NCSCR-1 was the first university-owned and operated reactor in the U.S.
Code: 0313 13 31201 624 711 81111 921 105
84677

No. 19 North Carolina State College Reactor-2

(NCSCR-2, NCSR-2, RRR-Raleigh Research Reactor)

Reference: IAEA Dir. of Nuc. Reactors, 2, 1959, pp. 189-98.
Originator: C. K. Beck.
Status: Critical May 1957; dismantled, December 1958. Modified to NCSCR-4.
Details: Modification of NCSCR-1. Differences: no coolant; cooling by natural convection. Core vessel: cylindrical with hemispherical bottom and flat circular top. Secondary cylindrical enclosure of Al. Control: 2 combined regulating and safety rods of stainless steel tubes containing tamped B_4C, with vertical movement. Power: 500 watts (+).
Code: 0313 13 31201 44 624 711 81111 921 101

No. 20 North Carolina State College Reactor-4

(NCSCR-4, NCSR-4, RRR-Raleigh Research Reactor)

References: NP-7041; IAEA Dir. of Nuc. Reactors, 2, 1959, pp. 189-98.

Originator: NCSC Reactor staff.

Status: In operation, June 30, 1961.

Details: NCSCR-2 moved to new location and modified. Main parts of NCSCR-2 used; new instrumentation, shielding, and gas recombiner installed. Power decreased to 100 watts (+).

Code: 0313 13 31201 44 624 711 81111 921 101

No. 21 Water Boiler

University of Utah

References: AECU-2900; Nucleonics 12, No. 4, p. 11, April 1954.

Originator: ?

Status: ? (Financing incomplete, 1954; designed.)

Details: Water-boiler-type reactor. Power: greater than 100 kw. Neutron flux: 3 or 4 x 10^{12} neutrons/cm^2/sec.

Code: 031X 13 31201 4X 624 7XX 8XXXX 9XX 101

No. 22 Alecto I Water Boiler

Centre d'Études Nucléaires de Saclay, Gif-sur-Yvette, France

Reference: D. Breton, letter to the editor, Nucleonics, 19, No. 1, p. 6, 1963.

Originator: Centre d'Études Nucléaires de Saclay staff.

Status: Became critical, 1961. Presumed still operating.

Details: Special-purpose testing reactor. Thermal neutrons, steady state, burner. Solution of uranyl nitrate (90% U^{235}) in H_2O. Various reflectors used. Maximum thermal flux, 10^8 neutrons/cm^2/sec. Power: few watts.

Code: 0313 13 31201 44 624 711 8XXXX 9XX 101

No. 23 Alecto II Water Boiler

Centre d'Études Nucléaires de Saclay, Gif-sur-Yvette, France

Reference: D. Breton, letter to the editor, Nucleonics, 19, No. 1, p. 6, 1963.

Originator: Centre d'Études Nucléaires de Saclay staff.

Status: Was to become critical, 1962-63.

Details: Same as Alecto I, except that plutonium nitrate solution is fuel.

Code: 0313 13 31201 46 624 711 8XXXX 9XX 101

No. 24 Tientsin Homogeneous Research Reactor

Nan-K'ai Polytechnic Institute (China)

References: AD-248402; CF-59-2-76.

Originator: Unknown; built and operated by teachers and students.

Status: Startup, 1959; in operation, 1960.

Details: Reactor is designed for research, instruction and production of isotopes. Maximum neutron flux outside reactor: 4.5×10^6 neutrons/cm^2/sec. Power level: 3 watts.

Code: 0313 13 31201 4X 62X 7XX 8XXXX 9XX 101

No. 25 Test Tube Reactor

LASL

References: Nucleonics, 13, No. 11, pp. 72-5, November 1955; Proc. 1st U.N. Int. Conf., 2, 1955, pp. 372-91.

Originators: R. P. Hammond and H. M. Busey.

Status: Conceptual stage, 1955; no further work.

Details: "Radical modification" of water boiler type. Thermal neutrons, steady state, burner. Fuel: highly enriched U^{235} as uranyl sulfate or phosphate in water. Fuel moves by convection up hot central tube and returns through surrounding annulus containing layers of cooling coils. Coolant: H_2O. Reflector: H_2O. Test-tube-shaped structure (cylindrical), lower portion containing critical core and heat exchanger and upper containing recombiner. Entire reactor contained in jacketed stainless-steel tube; all external pipes connected through cover. Assembly suspended vertically in a convection chimney. Controls: (1) negative temperature coefficient; (2) vertically moving curtain of boron steel; (3) sandwich of stainless steel or Dural containing samarium-gadolinium oxides; or (4) other neutron-absorbing materials. Power: 100 kw.

Code: 0313 13 31201 44 624 711 81144 921 105
84677

No. 26 Water Boiler

Babcock and Wilcox Company

Reference: TID-10117.

Originator: Babcock and Wilcox staff.

Status: Design stage, November 1955.

Details: Modified water boiler. Thermal neutrons, steady state, burner. Fuel: enriched uranyl sulfate in H_2O. Coolant: water circulating through tubes in solution; water is cooled in external heat exchanger. Reflector: 2 ft of graphite. Shielding: 4 in. steel thermal shield and 5 ft high-density concrete. Reactor housed in Zr or stainless-steel cylinder with hemispherical bottom. Negative pressure maintained to prevent escape of radioactive gases. Eleven Al-lined beam ports in faces of the shield for experiments. Control: negative temperature coefficient and use of two control rods. Power: 50 kw, with enough heat-transfer surface in core for higher power.

Code: 0313 13 31201 44 624 711 811XX 921 105
84677

No. 27 Homogeneous Reactor Gamma Irradiation Facility

ORNL

Reference: CF-56-7-126.

Originators: R. B. Briggs and J. O. Kolb.

Status: Preliminary design, made at request of Army Reactor Branch, July 27, 1956.

Details: Thermal neutrons, steady state, converter. Resembles HRT. Fuel-moderator-coolant: solution of uranyl sulfate in either H_2O or D_2O; 17 g U^{235}/liter in H_2O or 3.7 g U^{235}/liter in D_2O would be critical concentrations. $CuSO_4$ and excess H_2SO_4 added to either solution. High-pressure (1250 psia) circulating system in which reaction is sustained and low-pressure system in which fuels are stored, radiolytic gases are recombined, and fission-gases are separated for delivery to source. Reactor vessel: stainless-steel vessel, pear-shaped, 4 ft in diameter, with volume of 35 cu ft. Solution enters reactor at 440°F, 1500 gpm. It leaves at 525°F. Solution leaving reactor passes through gas separator, is cooled in heat exchangers, and pumped back to reactor. Steam produced at 420°F and 320 psia can be used to generate 3 Mw(e). Control: negative temperature coefficient and by alterating fuel concentration. Power: 15 Mw. Designed for irradiation of meat.

Code:								
0311	13	31201	44	624	711	84677	921	101
	14	31202		625		83779		

No. 28 Aqueous Homogeneous Research Reactor (HRR)

ORNL

References: ORNL-2256; TID-7540, pp. 288-301.

Originators: P. R. Kasten, M. I. Lundin, and C. L. Segaser.

Status: Feasibility study, 1957; project cancelled, 1961.

Details: Thermal neutrons, steady state, converter. Fuel: an 8% enriched uranyl sulfate (+ $CuSO_4$) solution in D_2O, the coolant-moderator. Solution enters reactor vessel through a diffusion screen at the bottom at 225°C, rises 50°C in temperature as it moves upward, and leaves at the top. Heat is transferred in 6 double-drum heat exchangers. Solution pressure: 1400 psia. Reactor vessel: 10 ft ID sphere of stainless-steel-lined carbon steel. No reflector. Control: negative temperature coefficient. Research facilities include six horizontal and one vertical experimental holes. A heterogeneous converter-plate design, constructed of seven thin concentric cylinders, intended to obtain high fast-neutron fluxes, may be used to enhance the over-all utility of the reactor.

Power: 500 Mw(t), 125 Mw(e).

Code: 0311 14 31202 42 625 743 84677 91X 101

No. 29 Homogeneous Advanced Engineering Test Reactor (AETR)

Aeronutronic Systems, Inc.

Reference: AECU-3478 (U-047), pp. 42-60; 85-103.

Originators: Staff of Aeronutronic Systems, Inc., subsidiary of Ford Motor Co., Glendale, California.

Status: Submitted in response to AEC request for recommendations of a concept for an AETR, March 29, 1957; no further work.

Details: Thermal neutrons, steady state, converter. Fuel-moderator-coolant is 10% enriched U^{235} as uranyl sulfate in D_2O. It is a bare, single-region reactor. Fuel solution is introduced into the core vessel at 208°F (98°C) at 4 points symmetrically distributed about the vertical axis in the lower hemisphere. Heated solution (308°F or 153°C) is withdrawn into the primary coolant circuit at four similarly distributed points in the upper hemisphere. Solution leaves the core through gas separators and passes through heat exchangers, from which it is returned to the core by the main circulating pumps. Core vessel: 8 ft ID stainless-steel sphere with external thermal shields 5 in. thick, around which is a cylindrical pressure vessel, 10 ft diameter. H_2O flowing through this vessel acts as a secondary coolant for the thermal shields. Seven experimental loop systems traverse the sphere vertically through the central region. Control: the negative temperature coefficient and by alteration of fuel concentration. Emergency shutdown could be accomplished by the injection of a boron solution into tubes in the core. Power: 500 Mw(t).

Code: 0311 14 31202 42 625 743 84677 921 101

No. 30 Homogeneous Nuclear Reactor with Subcritical Exterior Fuel Circulation Zone

Reference: Canadian Patent 614,567 (British No. 834,371).

Originators: F. T. Barr and J. F. Black.

Status: Patent granted, Feb. 14, 1961.

Details: Thermal (could be intermediate), steady state, burner. Fuel-coolant-moderator: 90% enriched U^{235} as uranium phosphate in H_2O. Horizontal core is surrounded by a graphite reflector, which may or may not be cooled. The solution is circulated through a flow path, through a reaction zone, and on to a cooling zone. Inlet temperature: 350°F; outlet: 650°F. So that the critical condition is not maintained throughout the conduit, sufficient neutron-absorbing material (Li, Cd, or B) is disposed circumferentially in the conduit leading into and out of the reaction zone to establish a subcritical exterior circulation zone. This neutron absorber may be horizontal rods or plates, clad with Al or stainless steel and located within the flowing solution. Reaction zone is 5-15 inches inside the closest absorbent. System pressure: 2000 psi; flow rate: 400 gal/sec. Control: 3 rods of Cd, B, or Li, moving vertically. This design is suitable for high-flux irradiation processing.

Code:	0313	13	31201	44	624	711	81111	921	101
	0212						81112		
							81116		

Variants: No. 1 -- Fuel may be a gas or vapor, such as UF_6 at 500-1000 psi.

Code:	0313	1X	31710	44	662	711	81111	921	101
							81112		
							81116		

No. 2 -- Fuel may be a compound--UF_4 or UI_4--that is liquid under the reactor conditions.

Code:	0311	1X	31XXX	44	613	711	81111	921	101
							81112		
							81116		

No. 3 -- Fuel could also be a solution of uranium in liquid Bi, uranium sulfates or nitrates, or uranium oxy-salts. Uranium or plutonium oxides could also be used.

Code:	0313	1X	31XXX	44	624	711	81111	921	101
					626		81112		
					634		81116		

No. 4 -- A fluidized solids system adapted to petroleum processing is another variation. Solid UO_2 particles, 75-200 μ, are conveyed by a gaseous coolant-moderator--hydrogen (or D_2O vapor). The fluid UO_2-H_2 is circulated similarly to the first design, but solids are separated in a cyclone and are later repressurized before they are returned to the system. Neutron-absorbing material at both ends of the reaction zone is B or Cd. The system is primarily for conversion, but can be adapted for breeding. UF_6 with solid particles of carbon suspended in it as moderator can also be used.

Code:								
0311	17	31915	4X	683	76X	81XX1	921	103
	14	31902		685		81XX2		
	12	31710		662				109

No. 31 Homogeneous Liquid Fuel Reactor

Aqueous Homogeneous Design

Reference: Canadian Patent 613,636.

Originator: H. J. Hibshman.

Status: Patent issued, Jan. 31, 1961.

Details: Several embodiments of the invention are described in the patent, which is aimed chiefly at methods by which to obtain higher neutron fluxes from homogeneous reactors by feeding fuel to and withdrawing it from the core through a plurality of inlet and outlet fuel streams.

Thermal neutrons, steady state, burner. Circulating fuel-moderator-coolant: solution of 93% enriched UO_2SO_4 (could be nitrate) in H_2O. Inlet temperature at the bottom of the core: 300°F; outlet at the top: 400°F. Operating pressure: 1000 psig. The core has 4 conduits which converge to form a quartered sphere, 1 ft in diameter. Shape may be cylindrical or rectangular. Each of the conduits, constructed of carbon steel, is circular (6 in. diameter) shaped outside the core and is altered to form one of the quarters inside the core. An alternate design describes cooling means for the conduits. Velocity of fuel: 6 ft/sec. Each conduit is provided with external heat exchanger. Reflector: Be and if desired, graphite. Control: negative temperature coefficient and fine or coarse Cd or B steel safety rods. Power: 28 Mw.

Two alternate arrangements are described. In the first, a plurality of pipes containing the fuel mixture converge in one critical zone and diverge outside of this zone. The fuel is fed in continuously through the 3 pipes and is passed on to the heat exchangers. The area of criticality may be defined by a solid moderator and control elements, to which structure this alternate arrangement is particularly applicable; or a liquid moderator (H_2O, D_2O, or hydrocarbons) may be circulated within pipe circuits, similar to those for fuel, through the critical zone, each circuit with cooling means.

A second arrangement comprises a fixed source of fuel within which a plurality of pipes pass liquid fuel continuously through the zone, to heat exchangers, and back to the core. The coolant (H_2O, D_2O, or hydrocarbons), which may also be the moderator, passes through the fixed fuel in pipe circuits. The coolant is also recirculated. Core material is stainless steel, and D_2O may be used with or in place of reflector materials listed for the main concept.

All of the above variants may be further altered by using a suspension of fuel instead of a solution.

Code:	0313	13	31201	44	624	711	84677	921	101
							81X11		
							81X12		

Variants: No. 1 -- The UO_2SO_4 could be dissolved in D_2O. Other features are the same.

Code:	0313	14	31202	44	625	711	84677	921	101
							81X11		
							81X12		

No. 2 -- A salt of U^{233} (the nitrate) could be dissolved in H_2O or D_2O.

Code:	0313	13	31201	45	624	711	84677	921	101
		14	31202		625		81X11		
							81X12		

No. 3 -- Plutonium-239 sulfate could be used as fuel in H_2O or D_2O.

Code:	0313	13	31201	46	624	711	84677	921	101
		14	31202		625		81X11		
							81X12		

No. 32 Homogeneous Advanced Engineering Test Reactor (AETR)

Adv. Scientific Techniques Research Assoc. (ASTRA)

Reference: NYO-4849.

Originators: R. G. Mallon, J. Saldick (ASTRA) and R. E. Gibbons (Catalytic Construction Co.).

Status: Submitted in response to AEC request for recommendations of a concept for an AETR, March 1, 1957.

Details: Thermal neutrons, steady state, burner. Fuel: hot, pressurized, acid solution of uranyl sulfate in D_2O, highly enriched in U^{235}. Moderator-reflector: 4 in. graphite blocks (sticks) with slots cut on each of the four faces into which graphite keys are inserted. The vertical cruciform openings between sticks are passages for the air coolant. The blocks are arranged in an octagonal shape, (7 ft high and 8 ft in diameter) supported on a steel plate, all of which is surrounded by a cylindrical vessel containing flowing demineralized water. It approximates a right circular cylinder with a vertical axis of graphite. Twenty-four Zircaloy-2 vertical fuel tubes, 5 in. diameter, are inserted into graphite structure from the bottom. Fuel flows upward in the fuel tubes and downward through a concentric flow separator in the center through gas separators to steam generators, and then back to the core, pumped by canned rotor pumps. Inlet temperature: 428°F; outlet: 479 to 503°F. Pressure: 1200 psia. Reflectors: top and side reflectors, 2 ft thick. Internal neutron shield of borated graphite is located just below the core. Control: negative temperature coefficient. Nine vertical through-holes for testing reactor fuel elements and assemblies at high temperatures and high thermal neutron fluxes. Power: 220 Mw(t).

Code: 0313 12 31714 44 625 711 84677 921 109

No. 33 High-Flux Research Reactor for Large-Volume Irradiations

Bendix Aviation Corporation

Reference: TID-2507 Del., pp. 329-334.

Originators: R. G. Mallon and W. R. Pearce.

Status: Conceptual design, June 3, 1955.

Details: Thermal neutrons, steady state, burner. Fuel-moderator-coolant: fully enriched solution of uranyl sulfate in H_2O, 30 grams per liter. There is also a graphite moderator, which is fabricated in 4 in. x 4 in. blocks. Reflector: 20-in.-thick borated graphite. Solution enters core vessel at 100°F and a velocity of 25 ft/sec and circulates through the graphite moderator in 24 parallel zirconium tubes. It leaves at an average temperature of 145°C to external heat exchangers. Because a conventional tube header would become critical, four manifolds at either end of the reactor and four separate heat exchangers are used. Core vessel: right-circular cylinder, 8 ft x 8 ft, enclosed in a concrete shield. Experimental facilities: fifteen removable graphite stringers--one of which permits irradiation of samples up to 16 in. x 16 in.--and up to forty-eight 4 in. x 4 in. beam tubes. Control: through the negative temperature coefficient, alteration of fuel concentration, and four vertical control rods. Design power: 120 Mw, average flux in the graphite moderator: 4.4×10^{14}.

Code:	0313	110	31201	44	624	71	84677	921	101
							83779		
							8111X		

References

1. "Nuclear Reactors Built, Being Built, or Planned in the United States as of December 31, 1963," TID-8200 (9th Rev.), USAEC, 1964.
2. National Research Council, Glossary of Terms in Nuclear Science and Technology, Am. Soc. of Mechanical Engineers, New York, 1957.
3. B. Lubarsky and D. J. Connolley, ed., "NACA Zero Power Reactor Facility Hazards Summary," NACA-RM-SE-57-F-28, Lewis Flight Propulsion Laboratory of NACA, Cleveland, Ohio, June 24, 1957.
4. AEC press release, March 23, 1962.
5. D. Fieno, "Consistent P1 Analysis of Aqueous Uranium-235 Critical Assemblies," NASA-TN-D-1102, NASA, Cleveland, O., November 1961.
6. A. D. Callihan, J. W. Morfitt, and J. T. Thomas, "Small Thermal Homogeneous Critical Assemblies," Proc. 1st U. N. Int. Conf. on Peaceful Uses of Atomic Energy, 5, pp. 145-55, United Nations, New York, 1956.
7. C. K. Beck, A. D. Callihan, J. W. Morfitt, and R. L. Murray, "Critical Mass Studies," Part III, K-343, Carbide and Chemicals Corporation, Oak Ridge, 1949.
8. J. R. Brown, B. H. Noordhoff, and W. O. Bateson, "Critical Experiments on a Highly Enriched Homogeneous Reactor," WAPD-128, Westinghouse Electric Corp., Atomic Power Division, Pittsburgh, May 1955.
9. J. Tachon, "Dispositifs de Transvasement de Solutions de Plutonium Pour une Expérience Critique," CEA-808, Centre d'Études Nucléaires de Saclay, 1958.
10. J. Bertrand, P. Bonnaure, C. Clouet d'Orval, J. Corpel, J. de Lamare, P. Lecoustey, I. Prevot, R. Roche, M. Sauve, J. Tachon, and G. Vendryes, "Proserpine, a Homogeneous Critical Experiment with Plutonium," Proc. 2nd U. N. Int. Conf. on Peaceful Uses of Atomic Energy, 12, pp. 539-562, United Nations, New York, 1958.
11. W. S. Hogan, "Homogeneous Reactors," in U. S. Research Reactors, Operation and Use, J. W. Chastain, Jr., ed., Addison-Wesley Publ. Co., Reading, Mass., 1958, pp. 111-46.
12. A. W. Kramer, Boiling Water Reactors, Addison-Wesley Publ. Co., Reading, Mass., 1958, p. 31.

13. "Water Boiler, Los Alamos Scientific Laboratory," LADC-442, (MDDC-1424), LASL, declassified Oct. 27, 1947.

14. R. F. Christy, "Theoretical Discussion of a Small Homogeneous Enriched Reactor," MDDC-72, Institute of Nuclear Studies, University of Chicago, Decl., June 18, 1946.

15. R. F. Christy, "Neutronic Reactor," U. S. Patent 2,843,543, July 15, 1958. Filed Oct. 19, 1945.

16. R. F. Wilson, J. O. Henrie, W. N. McElroy, W. E. Parkins, and J. W. Flora, "Aqueous Homogeneous Type Research Reactors," Proc. 2nd U. N. Int. Conf. on Peaceful Uses of Atomic Energy, 10, United Nations, New York, 1958. pp. 404-18.

17. J. A. Lane, H. G. McPherson, and Frank Maslan, Fluid Fuel Reactors, Addison-Wesley Publ. Co., Reading, Mass., 1958.

18. L. D. P. King, "Design and Description of Water Boiler Reactors," Proc. 1st U. N. Int. Conf. on the Peaceful Uses of Atomic Energy, 2, United Nations, New York, 1956, pp. 372-391.

19. Ref. 17, p. 347.

20. "U. S. Manufacturers Offer these Reactors for Export," Nucleonics, 14, pp. 70-75, November 1956.

21. "An Enriched Homogeneous Reactor," Rev. Sci. Instr., 22, No. 7, pp. 489-99, July 1951, (same as AECD-3059, LADC-887).

22. C. P. Baker, H. K. Daghlian, G. Friedlander, M. G. Holloway, D. W. Kerst, and R. E. Schreiber, "Water Boiler," AECD-3063 (LADC-819), LASL, Sept. 4, 1944, decl. Feb. 21, 1951.

23. Ref. 17, pp. 340-347.

24. F. L. Bentzen, R. E. Carter, J. Hinton, L. D. P. King, J. C. Nevenzel, R. E. Schreiber, J. W. Starner, and P. H. Watkins, "High-Power Water Boiler," AECD-3065 (LADC-822), LASL, Sept. 19, 1945, decl. Feb. 27, 1951.

25. L. D. P. King, "A Brief Description of the Los Alamos Homogeneous Reactor, SUPO Model of the Water Boiler," AECD-3287 (LADC-1081), LASL, October 1951, decl. Dec. 27, 1951.

26. L. D. P. King, "The Los Alamos Homogeneous Reactor, SUPO Model," LA-1301, LASL, Feb. 7, 1952.

27. L. D. P. King, "Water Boiler Reactor," U. S. Patent 2,961,391, Nov. 22, 1960.

28. E. A. Moore, Jr., "Operating Manual for the AE-6 Reactor," NAA-SR-MEMO-5395, Atomics International, June 23, 1960.

29. R. Chalker, "Design and Construction of Water Boiler Neutron Source," Science, 119, pp. 9-15, Jan. 1, 1954.

30. M. E. Remley, "Operation of a Water Boiler Neutron Source," NAA-SR-839, Atomics International, Nov. 20, 1953.

31. H. Pearlman, "Reactors for General Research Use," in Proc. Univ. Research Reactor Conf. Held at Oak Ridge, Tenn., Feb. 17-18, 1954, W. W. Grigorieff, ed., AECU-2900, Oak Ridge.

32. IAEA Direction of Nuclear Reactors, Volume II, Research Test and Experimental Reactors, International Atomic Energy Agency, Vienna, 1959.

33. A. T. Biehl, T. Fahrner, and S. Kash, "A Water-Boiler Reactor as a Source of Neutrons for Exponential Experiments," Nuclear Science and Technology (Extracts from Reactor Science and Technology, 2, Issues 1-4, April-December 1952), pp. 69-76, TID-2503, Del.

34. M. E. Remley, "Preliminary Outline of the Critical Assembly and Initial Operational Testing of L-3 Reactor," NAA-SR-MEMO-784, Atomics International, Sept. 25, 1953.

35. "Description of a Homogeneous Solution Type Research Reactor," NAA-AER-1023, North American Aviation, Inc., June 23, 1954.

36. Ref. 17, pp. 347-348.

37. "Starting Nuclear Sixties; Who's Building the Civilian Reactors," Nucleonics, 18, No. 1, p. 19, January 1960.

38. L. Reiffel , "The First Industrial Research Reactor Facility Design, Operational Experience, and Research Programs," Proc. 2nd U. N. Int. Conf. on Peaceful Uses of Atomic Energy, 10, pp. 393-403, United Nations, N.Y., 1958.

39. "Nuclear Reactor for Medical Research," North American Aviation, Inc., NAA-AER-1180, January 1955.

40. J. E. Gilligan, Jr., R. S. Hart, and O. D. Seawell, "Hazards Summary Report, L-77 Laboratory Reactor," AI Memo-2387, Atomics International, Jan. 20, 1958.

41. J. W. Flora, "Educational Uses of the Small 5-Watt Laboratory Reactor," Proc. 2nd Nuclear Eng. and Sci. Conf., 1957, Philadelphia, Advances in Nuclear Engineering, 2, p. 83, Amer. Soc. Mech. Engrs., Pergamon Press, 1957.

42. "Reactor Included in Beirut Exhibit," ANS Nuclear News, 4, No. 11, p. 26, November 1961.

43. "A 5-W Laboratory Reactor," Nuclear Eng., 1, No. 8, pp. 344-5, November 1956.

44. J. H. Lampe, "Nuclear Engineering Program at North Carolina State College," TID-7527, pp. 134-142, OTS, 1956.

45. C. K. Beck, A. C. Menius, Jr., R. L. Murray, Newton Underwood, A. W. Waltner, and George Webb, "Further Design Features of the Nuclear Reactor at North Carolina State College," AECU-1986, (NCSC-46), North Carolina State College, January 1952.

46. C. M. Baldwin, Jr., D. B. Beman, H. B. Carter, H. A. Lamonds, J. W. Meeks, A. C. Menius, Jr., R. L. Murray, E. J. Story, and A. W. Waltner, "Summary Hazards Report for the North Carolina State College Reactor NCSCR-4," NP-7041, North Carolina State College, Nov. 5, 1958.

47. A. C. Hughes, "University Reactor Conference," Nuclear Power, 5, No. 54, pp. 99-101, October 1960.

48. "The Role of Nuclear Reactors in University Research Programs," NSF-60-39, National Science Foundation, July 1960.

49. L. B. Borst and B. A. Mong, "The University of Utah Reactor Project," in Reference 31.

50. Nucleonics, 12, No. 4, p. 11, April 1954.

51. D. Breton, "More French Reactors," (letter to the editor), Nucleonics, 19, No. 1, p. 6, January 1963.

52. J. W. Ullmann, "Foreign Research and Power Reactor Preliminary List," CF-59-2-76, ORNL, Feb. 26, 1959.

53. "Reactors Developed in the USSR and Its Bloc Countries," AD-248402, ASTIA, Oct. 31, 1960, p. 43.

54. J. British Nuclear Energy Conference, 2, pp. 395-407, October 1957.

55. H. M. Busey and R. P. Hammond, "Test Tube Research Reactor," Nucleonics, 13, No. 11, November 1955, pp. 72-75; Ref. 18.

56. "Reactor Studies--Final Report. Part I," TID-10117, Babcock and Wilcox Co., November 1955, pp. 106-113.

57. E. D. Arnold and A. T. Gresky, "Exploratory Study: Homogeneous Reactors as Gamma Irradiation Sources," CF-56-6-107, ORNL, July 1956.

58. R. B. Briggs and J. O. Kolb, "A Homogeneous Reactor Gamma Irradiation Facility," CF-56-7-126, ORNL, July 27, 1956.
59. P. R. Kasten, M. I. Lundin, C. L. Segaser, R. E. Aven, D. R. Gilfillan, R. F. Hughes, M. C. Lawrence, H. A. McLain, R. A. McNees, C. Michelson, and C. W. Nestor, "Aqueous Homogeneous Research Reactor--Feasibility Study," ORNL-2256, ORNL, May 9, 1957.
60. W. A. Gall and M. I. Lundin, "Homogeneous Reactors. Design Section Progress Report for June 1956," CF-56-7-63, ORNL, July 13, 1956, pp. 9-13.
61. M. I. Lundin, "Homogeneous Research Reactor," in "HRP Civilian Power Reactor Conference Held at Oak Ridge National Laboratory, May 1-2, 1957," TID-7540, ORNL, July 1957, pp. 288-301.
62. "A Selection Study for an Advanced Engineering Test Reactor," AECU-3478 (U-047), Aeronutronic Systems, Inc., March 29, 1957.
63. F. T. Barr and J. F. Black, "Homogeneous Nuclear Reactor with Sub-critical Exterior Fuel Circulation Zone," Canadian Patent 614,567, Feb. 14, 1961.
64. H. J. Hibshman, "Homogeneous Nuclear Reactors," Canadian Patent No. 613,636, Jan. 31, 1961.
65. R. G. Mallon, J. Saldick, and R. E. Gibbons, "Conceptual Design of an Advanced Engineering Test Reactor," NYO-4849, (ASTRA-200-E-1.1), Advanced Scientific Techniques Research Associates, March 1, 1957.
66. R. G. Mallon and W. R. Pearce, "A High-Flux Research Reactor for Large-Volume Irradiation," Nuclear Science and Technology, 1A, Issue 2, pp. 329-334, August 1955, (TID-2507 Del.).
67. W. M. Breazeale, "Research Reactor Program at the Pennsylvania State University," Nuclear Engineering, Part II, Chem. Eng. Progress Symposium Series, 50, No. 12, pp. 6-10, Amer. Inst. Chem. Engrs., 1954.
68. J. A. Lane, "Where Reactor Development Stands Today," Nucleonics, 14, pp. 30-37, August 1956.
69. J. A. Lane, "Aqueous Fuel Reactors," Chapter 19 in Reactor Handbook, Vol. IV. Engineering, 2nd ed., Stuart McLain and J. H. Martens, eds., Interscience Division, John Wiley and Sons, New York, 1964.

70. H. H. Cappel, "Hazards Summary Report for the Walter Reed Army Medical Center Nuclear Research Reactor," AI-3739, Atomics International, May 6, 1959.

71. "Solution-Type Research Reactor Has Unique Feature," Atomics, p. 15, March-April 1963.

72. Atomics, p. 4, March-April 1963.

Additional References

M. Monet, "Homogeneous Pile Water Activity," Final Report, MonE-64, ORNL, Feb. 8, 1946, decl. Jan. 30, 1956, p. 13.

K. A. Hub, "Discussion of Homogeneous Reactor Possibilities," AECU-3336 (IC-KAH-56-5), Internuclear Corp., May 16, 1956.

"Research Reactors," Nuclear Engineering, 5, No. 46, pp. 99-104, March 1960.

"World Reactor Chart, Third Edition," Nuclear Power, 1, No. 69, January 1962.

F. E. Croxton, "List of References on Homogeneous Reactors," TID-299, USAEC, Oak Ridge, Jan. 31, 1955.

Musa Halev, "Survey of Reactor Facilities for Radiation Biology Research," NADC-MA-5813, U. S. Naval Air Development Center, Oct. 22, 1958.

[illegible] Nuclear Research [illegible] 1969, [illegible]

[illegible] 1964.

[illegible]

[illegible], [illegible]

Additional References

[illegible] Final Report, [illegible]

[illegible]

[illegible] Possibilities," [illegible]

[illegible] 1964.

[illegible] Nuclear [illegible]

[illegible] Nucleon [illegible] 1964.

[illegible] Process and [illegible]

[illegible]

[illegible] Facilities for Radiation [illegible]

[illegible]

Chapter 3 Non-Boiling Reactors for Power and Breeding

The reactor concepts in this chapter include those in which a homogeneous aqueous solution or slurry (suspension) is utilized as fuel. These fuels were among the first to be suggested, but reactors employing them have not been developed beyond the prototype stage. An excellent description of aqueous solution and slurry reactors is given by Lane, MacPherson, and Maslan.[1] Also valuable are reviews by Briggs and Swartout[2] and by Brown and Morris.[3]

These reactors can be classified in many ways--purpose, method of heat removal, origin, materials used, etc. Table I (Chapter 1) lists the different types and their applications.[4] A useful classification is by fuel utilization--burners, converters, or breeders. Breeders are either one-region--consisting of a homogeneous solution or slurry; or two-region--a core plus a blanket of fertile material. In the course of the Homogeneous Reactor Experiment series, the Homogeneous Reactor Project, and the Los Alamos Power Reactor Experiments, groups of closely related concepts were developed. Thus, they are discussed together, rather than under their respective classifications. Other homogeneous aqueous solution reactors are then grouped as burners, converters, or breeders. The closely related concepts of non-homogeneous aqueous reactors and colloidal fuels are briefly discussed, as there is little in the literature concerning them. Slurry-fueled reactors are treated separately in this chapter because of their special features. For boiling homogeneous aqueous reactors, see Chapter 4.

Early History

It is hard to determine who proposed the first homogeneous aqueous reactor for power and breeding. Early in the Manhattan Project, aqueous homogeneous solutions and slurries were considered as fuels for producing plutonium but, as with research reactors, the need for enriched fuel and large quantities of heavy water made heterogeneous reactors preferable at the time. When enough of heavy water and enriched fuel became available, aqueous solutions and slurries were studied for breeding and power production. A group of chemists at the Clinton Laboratories, now the Oak Ridge National Laboratory, were active in planning homogeneous reactors for research and breeding. In 1944, Nordheim reported on the potentialities of a breeder reactor.[5] In an extensive experimental program, research workers studied the choice of fuel, moderator, and cooling mechanism; problems of corrosion, poisoning, and formation of radiolytic gases; and most of the other aspects of reactor development.

Investigative work continued through 1945. At that time, such difficulties as bubble formation, corrosion, solution instability, and external holdup of fissionable material led to the decision to consider heterogeneous reactors instead.[6] The early homogeneous aqueous reactor designs are discussed under their respective classifications, and details are given on the data sheets. Many of them utilized a heavy water solution of sodium uranyl carbonate or uranyl sulfate as fuel. Thorium metal or oxide was a common fertile material. They were cooled by circulating the fuel solution to external heat exchangers.[7,8]

Major experimental efforts on power reactors apparently began in 1950 at Oak Ridge National Laboratory. At that time, members of the staff began the design and building of a pilot plant for a fluid-fuel reactor, Homogeneous Reactor Experiment No. 1, (HRE-1). Most of the later interest in homogeneous aqueous solution reactors arose after the successful operation of HRE-1, which removed most of the previous concern that difficulties in operation and limitations on power production would make such reactors impossible. In spite of the many new reactor concepts based on the HRE experiments that developed, HRE-1 and HRE-2 were the only reactors built in the series.

HRE-1 and Related Concepts

Before the first Homogeneous Reactor Experiment (HRE-1), preliminary pilot designs, were proposed and, after the success of HRE-1, a series of related concepts followed.

HRE-1 came into being partly owing to the doubt that a high-powered aqueous homogeneous reactor could be operated at all because of the high gas production and consequent bubbling. The purpose was to investigate circulating uranium solution at high enough power and temperature to produce electricity from the heat released. Some of the nuclear and chemical problems studied included not only the bubbling problem, but also the corrosion of the reactor-core materials and the tendency of the fuel solution to become unstable, causing either precipitation of hydrolysis products from the fuel or formation of two fuel phases, one being an extremely corrosive heavy liquid. Varying core arrangements and supercharging with oxygen were considered as means of reducing corrosion, and adding sulfuric acid made the fuel stable at higher temperatures.[9]

Preliminary Work

Weinberg has described 20-kw(e) pilot model for designing larger homogeneous reactors.[10] It was suggested that the long-range homogeneous reactors would probably be moderated with heavy water but, since ordinary water has a much smaller slowing-down length than heavy water, substitution of ordinary water for heavy water would produce a scaled-down reactor. This scaled-down model would reproduce all the essential features of the full-scale plant; hence it would be ideal as a pilot plant. The pilot model was to have 2 kilograms of 93.4 percent enriched uranyl sulfate in ordinary water solution, and the reacting solution was to be pumped through the reactor vessel and heat exchanger. High temperature (400°F) and pressure (1000 psi) would be sufficient to keep the bubble volume manageably small and to give a 10 percent over-all power efficiency. A heavy-water blanket as a reflector under 1000 psi surround the reactor.

The HRE Project

A full-scale reactor was not used at the start but rather a reactor experiment, which is defined as:[11]

> .."a reactor in the research and development program, usually producing less than 10 Mwt, designed for the limited purpose of testing the technical feasibility of a reactor concept or some unique reactor feature or piece of equipment. A reactor experiment is built with the intention of making changes in fueling structure, associated components, or other components of the reactors."

According to another definition:[12]

> "A reactor experiment makes no attempt to incorporate features which would be a necessary part of any full-scale reactor. Reactor experiments have, as their purpose, the testing of an idea or a unique piece of equipment. As a matter of fact, a reactor experiment is just one phase of the research and development

program and should be considered as such. The size is immaterial, but, generally, reactor experiments will be small with generation capability of less than 10 Mw. Indeed, the actual generation of electricity is unnecessary unless it is a vital part of the experiment."

It was decided to build, at Oak Ridge, HRE-1, a 1000-kw, uranyl sulfate reactor. The primary purpose was to determine whether stable operation at the 20-kw/liter level was possible at all; all other considerations, like ease of maintenance, corrosion, and chemical reprocessing, were almost ignored because it was believed that the primary question could be answered quickly, after which the experiment could be dismantled. The core of HRE-1 was a Type-347-stainless-steel-sphere, 18 inches in diameter, through which circulated a 0.17-molar highly enriched solution of uranyl sulfate in ordinary water at 1000 psi, with an outlet temperature of 250°C. The core was surrounded by a 10-inch-thick blanket of heavy water.[13-15]

HRE-1 achieved its full power rating of 1000 kw on February 23, 1953, becoming the second reactor to produce electricity. It reached a maximum power of 1600 kw on January 29, 1954. It was dismantled two years after it was built, largely because it had served its primary purpose and there were no special provisions for maintaining it. It demonstrated that a high-powered aqueous homogeneous reactor could be run stably, and it resolved the original question whether gas production would make aqueous reactors _a priori_ unworkable.[16] The HRE-1 concept, with heavy water possibly substituted for ordinary water, was considered for submarine propulsion.[17] For details on the design of HRE-1, see reference 18.

This experience led to a more ambitious experiment, HRE-2, also known as HRT, the Homogeneous Reactor Test.[19] It first achieved high-power operation on February 8, 1958.[13] HRE-2 had a Zircaloy core vessel and the intention was to try thoria slurries as well as heavy water in the reflector.[20] During March and April 1958, while the reactor was being operated at high power (maximum power of 6.3 Mw), a hole was melted in the core tank after 30 minutes of operation at 6.3 Mw. The hole resulted from an unsatisfactory hydrodynamic design of the core and the high-temperature instability of the fuel. In addition to the hole, other damage included severe local overheating of some metal surfaces. Hence, more emphasis was placed on core design and hydrodynamics studies in the development programs.[21] See also references 22 and 23.

The core was inspected and, because the hole was inaccessible, nuclear operation was resumed; core and blanket were operated together as a one-region reactor. The core was repaired by patching in late 1960; by this time there were two holes in the core vessel. HRE-2 attained a power level of 5 Mw on January 4, 1961 with no indication of the chemical instability previously noted. However, work on HRE-2 ended when the AEC terminated the entire homogeneous program on June 30, 1961. After the reactor was shut down, inspection revealed that one of the core patches had failed.[24,25]

After HRE-2 was shut down and before the Homogeneous Reactor Project was terminated, several preliminary or conceptual design studies were made of a 5-10 Mw experimental replacement reactor suitable for installation in the existing HRE-2 facility with a minimum of system changes. This design included a cylindrical core with a beryllium reflector surrounded by a blanket of thorium pellets enclosed in a removable steel pressure vessel. The fuel was a solution of 6.56 grams of uranium-233 per liter.[26-28]

HRE-1 and HRE-2/HRT were the only reactors to reach the hardware stage, unless the HRT mockup be also considered a reactor. Seven other reactors, however, were conceived in the pattern of HRE-1. In order of increasing size, the nine are: HRE-1, HRE-2 as conceived, HTR (the Homogeneous Thorium Reactor), HRE-2/HRT as built, HRT as conceived, HRE-3, HRE-4, ISHR (the Intermediate-Scale Reactor), and TBR (the Thorium Breeder Reactor). These concepts and closely related ones have been discussed in several reports.[29-33]

HRE-2 and HRT as conceived are related to the HRE-2/HRT actually built (hereafter referred to as HRT): The development program for homogeneous reactors as set forth in the original "Five Year Plan" by the AEC called for two intermediate development steps after the HRE-1: the HRE-2 and the HRT. It was decided, however, to extrapolate the HRE-2 to an intermediate size, with a more-flexible experimental plant, compromising between its original size and that for the larger HRT.[34]

The HRT design has been reviewed.[35] The distinction between HRE, HRT, and TBR (the Thorium Breeder Reactor) can be clarified by analogy with the fast-breeder program. The Homogeneous Reactor Program resembled the fast-breeder program in its general aspect of building a succession of larger reactors ending in the "ultimate" full-scale reactor. Analogous to the EBR, EBR-2, PBR sequence are the HRE, HRT, TBR. The final stage of the Homogeneous Reactor Program would involve a full-scale breeder reactor.[36]

The HTR, or Homogeneous Thorium Reactor, preceded the HRT as a concept. It was judged unnecessary to carry this intermediate-scale reactor to the hardware stage, and it was therefore superseded by the Homogeneous Reactor Test, HRT. The Homogeneous Thorium Reactor was planned to produce 65,000 kw(t) of which 16,000 kw would be converted to electricity. A blanket of thorium, from which uranium-233 would be produced, was planned. The core would not be as large as that for a full-scale reactor but the thickness of thorium blanket and concentration of fertile material would be the same as for a central power station of this type.[37]

The third phase of the homogeneous program--presumably after HRE-1 and HRE-2 were conceived--was the medium-sized Homogeneous Thorium Reactor (HTR), a thorium breeder fueled with uranium-233 to produce 65,000 kw(t).[37,38] The HRE-2 concept was later modified to be more useful and more powerful than HRE-1, with power levels of 10 megawatts rather than 3 megawatts planned. It was more versatile, and by 1957 included a blanket system with chemical-reprocessing equipment as part of the reactor. Thus it was judged unnecessary to construct an intermediate-scale homogeneous reactor, because the modified HRE-2 would provide most of the technological information originally expected from the intermediate-size reactor.[39]

At an earlier date, a stage intermediate between HRT and TBR was considered. Research and design leading to construction of an intermediate-scale homogeneous reactor was undertaken. The reactor was conceived of as operating at a maximum of 40 Mw and to be of a size approaching that of a full-scale reactor, i.e., with a core tank 8 to 10 feet in diameter, compared to a diameter of 15 to 20 feet for a full-scale reactor.[40]

This Intermediate Scale Homogeneous Reactor (ISHR)[41,42] went through several variations. Its design was started in 1951-1952 to provide the information necessary for reasonable extrapolation from the HRE to a full-scale production reactor. Several types of blankets were considered for the ISHR, especially fluidized thorium oxide.[43] Further work was deferred late in 1953, however, when it became evident from HRE-1 and the associated development program that construction of a second homogeneous reactor experiment would be more suitable.[44]

Thus the ISHR, like HTR, was replaced by the HRT program, and the sequence ran from HRE to HRT and TBR.

ISHR was intended to demonstrate nuclear, chemical, engineering, and economic feasibility of an aqueous homogeneous solution reactor by actual testing on a scale smaller than that of the final full-scale reactor.[45] Some of the problems inherent in the power reactor but not apparent in the HRE-1 or HRT might be expected to appear at this point.

The problem of core construction, for example, was considered with the view of assuring feasibility of and economy in construction, stability of flow, adequate heat removal from all points, low pressure drop (preferably less than 15 psi), low gas holdup, and velocities near the walls consistent with corrosion problems. Various core configurations were considered, including straight-through and spherical shapes, but no specific choice was made prior to 1953, the end of the effort on ISHR.[46]

Designs based on the ISHR include the Cooling Tower Reactor,[47] a two-region converter,[48,49] a one-region breeder,[50] a two-region breeder,[51] and a three-region lithium converter.[52]

The Thorium Breeder Reactor (TBR), was intended to develop the concept of a thorium-uranium-233 breeder power station in sufficient detail to provide the basis for detailed design by an architect-engineer.[2]

One design reported[53] was for a power station, consisting of three reactors, each delivering 100 Mw. They are spherical, two-region reactors, with enriched uranyl sulfate solution in heavy water as fuel and a slurry of thorium oxide in heavy water as fertile material.

In 1955, it was decided that, because of insufficient data, intelligent study of a specific reactor was not possible and the original program was modified to include a preliminary investigation of four variants of the thorium breeder reactor, in the hope that advances in technology would permit selection of one of them for further study later.[54] The TBR design, however, was set aside because of urgent work needed for the HRT.[53]

HRE-3 was designed to provide operational and technical data and to demonstrate technical feasibility of an intermediate-scale, two-region aqueous homogeneous power breeder, but it was interrupted to apply additional effort to problems encountered in slurry-fueled reactors, and, to await resolution of problems that had arisen in operation of the HRT.[55,56] HRE-3 had been intended to study the two-region power concept on a small scale, starting in 1961 with a 50-Mw prototype plant, with further development depending on the success of the prototype.[57]

In HRE-3, the fuel was uranyl sulfate solution in heavy water, with a blanket slurry of thorium oxide in heavy water, a 50-Mw core, a 10-Mw blanket, and an electrical output of 19 Mw.[55]

HRE-4 was a slurry-fueled reactor, which will be described under that category.

The Homogeneous Reactor Project

The ORNL Homogeneous Reactor Experiment Project (HRE) was reconstituted in about 1951 as the Homogeneous Reactor Project (HRP) to study the feasibility of a full-scale homogeneous reactor to produce plutonium.[58] Many alternative designs were studied, and attention was given to such problems as the behavior of solutions and slurries in reactors and corrosion difficulties. For example, in 1956, members of the project reported on HRT and five other possible designs: a U^{235} burner fueled with uranyl sulfate in heavy water; two one-region U^{238}-Pu converters, one fueled with uranyl sulfate solution, the other with a slurry of uranium trioxide and plutonium dioxide in heavy water; and two Th-U^{233} breeders, a two-region and a one-region, one fueled with solution, the other with slurry.[59] Also a breeder with a blanket of thorium oxide pellets was later suggested.[33] Designs for two other solution-fueled reactors were for power reactors producing 380 Mw(e).[60,61] The project, along with the entire subject of fluid-fuel reactors, was later evaluated, as described under "Evaluation and Status."

A typical aqueous homogeneous reactor concept developed for evaluation purposes is the Aqueous Homogeneous Breeder Reactor.[62] This concept resembled many of the thorium breeders, with its aqueous uranyl sulfate solution as fuel and a slurry blanket of ThO_2 pellets. It especially resembles the reactor described in reference 33. Special provisions were made in the design of the blanket structure. A variation, in which thorium tetrafluoride pellets replace the thorium oxide in the blanket, was suggested to obtain higher fuel yields and lower fuel-cycle costs. Much of the claimed lower cost would be due to less-expensive fuel reprocessing.[63]

Boiling was considered as a means to remove power from the thorium oxide blanket of large two-region breeders like those studied by HRP, but it was rejected in favor of circulating the slurry.[64]

The LAPRE and Related Reactors

Two power-reactor experiments, somewhat analogous to HRE-1 and HRE-2, were carried out at the Los Alamos Scientific Laboratory. In Los Alamos Power Reactor Experiments I and II (LAPRE-I and LAPRE-II), the fuel solution consisted of uranium oxides dissolved in aqueous phosphoric acid.[65-68]

Phosphoric acid solutions of uranium have several advantages that make them attractive as reactor fuels. Their low vapor pressure and high thermal stability permit operations at higher temperatures than are possible with other aqueous solutions. Still other advantages are the high hydrogen density and low cross-section for neutron absorption, which permit the use of small, compact reactors and low fuel inventories. There is, however, a grave disadvantage. The extreme corrosivity of the acid solutions makes the choice of containment materials very difficult and requires unusual precautions. LAPRE-I and LAPRE-II were designed to determine whether such solutions were practical reactor fuels and to design reactors that could best use them.

The cylindrical form of the LAPRE reactor vessels has led to the name "Test-Tube Reactors" being applied to them.

LAPRE-I

LAPRE-I (originally designated DIR-P) was intended to be a compact, high-temperature nuclear steam generator of simple and compact design, in which the reactor and heat exchanger would be in a single vessel. Thus, external circulation of highly radioactive solution is avoided. The fuel solution was a solution of highly enriched uranium trioxide (111 grams U^{235} per liter) in aqueous phosphoric acid containing 53 wt.% acid. The cylindrical reactor had a critical zone above the cooling coils over which solution was circulated by a pump. Oxygen overpressure was used. Below the coils was a reservoir region, which contained a boron rod to prevent criticality and permitted storage of excess solution. To minimize corrosion, the vessel and all components exposed to the acid were gold-plated, except for certain items made of or coated with platinum.[66,67]

The reactor went critical in February 1956. A maximum operating temperature of 390°C at about 150 kw was achieved before a failure in the heat-exchange system ended the test. Inspection showed that the gold cladding on one of the exchanger tubes had ruptured. No more tests were made and the reactor was dismantled in January 1957. The experience, however, provided valuable information for the design of LAPRE-II.[66,67]

LAPRE-II

LAPRE-II differed in several ways from LAPRE-I. A more concentrated solution of phosphoric acid was used--95 wt.% instead of 53. The vapor pressure of the resulting solution, even at 450°C, was only about 800 psi, compared with 4500 psi for the LAPRE-I solution. Because of its reducing properties, this solution attacks only slowly any metal below hydrogen in the electromotive series. The lower vapor pressure of the LAPRE-II solution permits a reactor vessel of thinner walls. Instead of pumping the solution for cooling, circulation over the cooling coils was by natural convection. An external annular reflector of graphite and beryllium was provided. Gold cladding was again applied to exposed surfaces.[68,69]

Operation of this experiment began in February 1959 and continued into May of the same year. The maximum fuel temperature measured was 826°F. Full-power operation was reached on April 22. The maximum power was 745 kw, at 745°F and 670 psi. The operation was satisfactory at widely varying power demands, and the power was limited only by the capacity of the primary cooling loop. The gold cladding was shown to be a possible answer to the corrosion difficulty, but improvements in materials and fabrication methods are needed. As an advanced concept, the use of impervious graphite as container material was suggested, and in one reactor concept, such graphite would be material for the common container and primary heat exchanger.[68,69]

A version of LAPRE-II, with a spherical core, increased power, and other modifications were proposed for application to the DEW line.[70]

Sandia Phosphoric Acid Reactor

A model employing the LAPRE concept has been designed by Sandia Corporation for remote locations or mobile units.[71] It would generate from 100 to 1500 kw(e).

Other Homogeneous Aqueous Solution Reactors

Many concepts for aqueous reactors were developed after the success of HRE-1, and some were developed even before the completion on this experiment. A few were one-region converters or breeders, but most were one-region burners or two-region breeders.

One-Region Burners

One-region burners have received considerable attention as sources of industrial power. Published concepts are similar in several ways. In nearly all, for example, highly enriched uranyl sulfate solution in water is the fuel, and a system pressure of 1000-2000 psi is usual.

Much of the earlier work was on mobile reactors.

A reactor to produce 250 Mw(t) was described in a 1950 report by workers at ORNL.[72] It utilized enriched uranyl sulfate in light water and superheating steam by burning radiolytic gas, and it is otherwise similar to most other burners. Two variations were proposed. Each differed from the original only in such details as fuel enrichment and concentration.

Solution power burners have been considered for locomotion. A study at the University of Utah found that a locomotive powered by a nuclear reactor would be technically feasible.[73] The reactor would produce 30,000 kw as steam at 250 psig to operate a 7200-hp locomotive. The reactor contains a core of fully enriched uranyl sulfate solution within two hexagonal slabs. Adjacent to each slab is a water reflector connected to heat-transfer tubes, which penetrate the core. Water circulates through tubes and reflectors for cooling. Control is by means of rods in the core. Likewise, a Russian design for a locomotive, powered by a reactor fueled with a water solution of a uranium salt, has been described.[74] The economic feasibility of nuclear-powered locomotives has, however, been questioned.[75] A Wright-Patterson study of 1959, concluded that, for aircraft and remote applications, in general the problems of aqueous homogeneous reactors were not immediately surmountable.[76]

Students at the Oak Ridge School of Reactor Technology reported in a term paper a design for a solution burner for ship propulsion.[77] In the generally conventional design, a spherical core with concentric inlet and outlet was included because its flow pattern would be stable under the rolling conditions aboard ship.

Three other reactor concepts, for producing power in remote locations, developed by OSORT students were: a 10-Mw reactor;[78] "STUPO"--so named because of an unsuccessful attempt to extrapolate the data of LOPO and SUPO;[79] and an 80-Mw reactor.[80] The first reactor is claimed to be unique in that construction of a prototype is feasible without additional development. The STUPO project considered a reactor with a maximum power of 10 Mw and reliability for use in remote locations. The study concluded that feasibility depended on whether 10 Mw of heat could be removed by natural convection. Design power output of the 80-Mw reactor is 20 Mw(e), with an 80 percent plant factor.

One-region burners include such utility types as the Wolverine Electric Cooperative Reactor and those proposed by General Electric and Westinghouse.

The Foster Wheeler Company proposed in 1955 to construct an aqueous solution reactor for Wolverine,[81] but the project was cancelled in May 1958 because new projections showed an excessive increase in cost. In December 1957 engineers from Oak Ridge, Sargent and Lundy, and Foster-Wheeler redesigned the reactor, using highly enriched uranyl sulfate in heavy water, under pressure to prevent boiling.[82] Uranium-235 would have to be added periodically to the reactor because of depletion of the original fuel by fission.[83]

The General Electric 100 kw(e) power plant had a circulating fuel solution of highly enriched uranyl sulfate in light water contained in a titanium-lined carbon-steel pressure vessel, which also served as a reflector.[84] The vessel was cylindrical, with a hemispherical bottom. Control was by the temperature coefficient of reactivity of the core. The proposal included a standard straight tube-and-shell heat exchanger. Many of the details of this design were based on data from the Oak Ridge Homogeneous Reactor Project. Westinghouse described a one-region 80,000 kw uranium-235 reactor, in which the fuel was a dilute uranyl sulfate solution.[85] Corrosion was the major consideration in selecting most of the design parameters. Another Westinghouse concept of this type is described in reference 86.

Babcock and Wilcox considered both a non-boiling and a boiling solution burner.[87] The low-power non-boiling system consists of a uranyl sulfate solution in a cylindrical vessel under an oxygen pressure of 1200 psi. Heat exchange is through bayonet cooling tubes immersed in the solution.

Kasten and Claiborne[88] studied fuel costs for spherical one-region power reactors containing 90 percent enriched uranyl sulfate in either heavy or light water and producing 25 Mw(e). The feed is also 90 percent enriched. They concluded that the technology of this type was fairly well developed as a result of the homogeneous-reactor program, and in fact that the technology seemed sufficiently developed to warrant serious consideration of the construction of these burner reactors.

One-Region Converters and Breeders

Some of the earliest concepts developed under the Manhattan Project were of this type, usually with uranyl sulfate solution as fuel, with heavy water as the moderator.[89] A one-region breeder similar to HRE-1 was the Homogeneous Power Reactor.[90] It was fueled by a solution of plutonium-239 in heavy water, but few other details are available.

In 1950, the ORNL Long-Range Planning Group proposed a reactor for power and plutonium production.[91] A solution of enriched uranyl sulfate in heavy water flows through a spherical reactor into high-pressure tubing, through the reactor shield, and into heat exchangers.

In 1952, the Large Scale Homogeneous Reactor concept was published at Oak Ridge.[92] In this reactor, designed for plutonium production, the fuel was slightly enriched uranyl sulfate in heavy water, under a pressure of 1000 psia. A similar reactor has also been proposed.[93] Corrosion was the chief difficulty with both.

A study of the economics of one-region breeders compared with two-region breeders[94] showed that, for a three-reactor power station producing 375 Mw(e), the cost of power from a one-region reactor station would be 0.9 mills per kwh higher than that for the two-region station--7.1 mills per kwh vs. 6.2 mills per kwh. In another study,[95] one-region solution reactors were compared with slurry reactors. A one-region spherical solution reactor operating at 330°C and delivering 300 Mw(e) was found to be competitive with slurry-fueled reactors.

A Russian reactor that is apparently a one-region converter has been described in surveys of Russian reactors.[96,97] In this homogeneous reactor, designed as a prototype of a power station with high output, uranium enriched to 0.9 percent U^{235}, apparently as an oxide suspension in heavy water, is the fuel. The pressure of the coolant is from 20 to 50 atmospheres, and the maximum temperature is 260°C. The power is from 1,150 to 2,000 Mw(t) or 280-500 Mw(e). In the design stage in 1960, the reactor project was expected to require seven to eight years. For this period, the consumption of natural uranium was estimated to be 306 tons, and of 0.95 percent enriched uranium, 43 tons.

Two-Region Converters and Breeders

Two-region solution reactors, some of which generally resemble the Thorium Breeder Reactor of the HRE series, have received much attention and many concepts have been published.

Several concepts were developed by workers in the Manhattan Project.[98,99] In three of them, the fuel is enriched sodium uranyl carbonate dissolved in heavy water. The fertile material is thorium metal, either in a slurry or as rods. One, the Heavy Water Moderated U^{233} Pile, was designed to produce 100 Mw. A few others, generally similar in basic concepts, have also been described.[100,101]

Wigner has described an unusual early concept.[102] The core tank is a U-tube, with cooling tubes that extend into it from both sides, with the bottom portion free of these tubes. Pistons in each leg of the tube move the solution from left to right alternately so that it comes in contact with the cooling tubes. The U-tube contains a solution of a U^{233} salt in heavy water, and a slurry blanket of thorium in heavy water surrounds the bottom of the tube.

After World War II, and even before the completion of the HRE series, work at Oak Ridge resulted in concepts designed for both breeding and for electrical power.

An early power-breeder concept was developed at ORSORT. In the reactor, a solution of U^{233} as uranyl sulfate in light water is the fuel and a slurry of thorium tetrafluoride is the fertile material.[103] Both core and blanket are under 1000 psi. Students at ORSORT also developed a concept for a reactor to produce both power and special isotopes.[104]

Two more ORSORT concepts were the Ultimate Homogeneous Reactor[105] and the High Temperature Homogeneous Reactor.[106] The first was perhaps so named because its purpose was to investigate the ultimate limitations of aqueous homogeneous reactor technology and to assess, by means of a design example, how far the homogeneous line of development could be followed. After considering other types of reactors, the authors decided on uranyl sulfate fuel; heavy water as coolant-moderator, with natural circulation; and a blanket of a slurry of thorium dioxide in heavy water. These conditions were chosen as best for higher core-power density, higher thermal efficiency, and constant turbine steam conditions from no load to full load in a 200-Mw breeder. The High-Temperature Homogeneous Reactor differed especially in the fuel solution, in which uranium-233 trioxide and chromium trioxide form a solution in light water. The pressure is also higher than usual--3500 psi. This high pressure permits a higher outlet temperature of the solution. The design was for a plant using two such reactors to produce a total of 290 Mw(e).

Work at ORNL produced concepts for power and breeding that had some features that differed considerably from the earliest ideas. Visner, for example proposed a two-region reactor without a core tank.[107] The uranyl sulfate solution fuel is in the same container as the thorium oxide slurry blanket. They are kept separate by operating the reactor as a centrifugal separator, so that the solution remains in the center and the slurry at the periphery. In the K-49 reactor,[108] a solution of uranyl sulfate in heavy water, rather than thorium metal or slurry, is the fertile material in the blanket. A long-range concept developed at ORNL was the Thermal Power Breeder.[109] It is a two-region breeder, in which uranium-233 as uranyl sulfate is the fuel and a suspension of thorium oxide in heavy water is the fertile material. This reactor is designed for steam production in a power plant, with the additional feature of breeding more uranium-233.

Two other ORNL concepts aimed at power for commercial use were the Project Dynamo concept and a reactor for central station power. Engineering details for the first were developed by the Project Dynamo staff from an ORNL concept.[110] Three such reactors, producing 450 Mw(t) each, were designed for a power station. The fuel is a solution of uranyl (uranium-233) sulfate in heavy water and the blanket a slurry of thorium oxide in heavy water. One design for the central station power reactor utilized a solution of uranyl sulfate in heavy water as fuel and a slurry of thorium dioxide and uranium trioxide, also in heavy water, as fertile material.[111] Power produced would be 100 Mw(e).

A two-region reactor developed at Oak Ridge was aimed at plutonium production.[112] The blanket is a concentrated solution of natural uranyl sulfate in heavy water. Plutonium of low plutonium-240 content could be obtained by continuously processing the blanket solution. This reactor experiment was planned to simulate reactor operations.

The Nuclear Power Group, composed of several private companies, developed concepts for two-region power-breeder reactors. The group was formed in October 1953 for broadening the scope of the previous nuclear power studies by the individual companies--American Gas and Electric Service Corporation, Bechtel Corporation, Commonwealth Edison Company, Pacific Gas and Electric Company, and Union Electric Company of Missouri. The initial objective was to select a reactor design that might be constructed in the reasonably near future primarily to produce electric power.[113]

Although the pressurized-water reactor seemed to meet their requirements most closely, the Group conceived at least two homogeneous aqueous high-power, blanket breeder concepts. One design consisted of a spherical core--6 feet in diameter--of uranyl sulfate dissolved in heavy water. The core is surrounded by a spherical blanket--2 feet thick--of thorium oxide and uranium slurry in heavy water.[114] Uranium-235 could be used to start the reactor and it could also be adapted to the uranium-238 - plutonium cycle. The concentration, breeding gain, and reactor calculations were based on Oak Ridge data,[115-118] and they were extrapolated to determine concentration and power gain for this NPG reactor. Staff members at Oak Ridge also aided NPG in completing this design, which is much like Oak Ridge designs.

By July 1957, the Nuclear Power Group, which now also included Central Illinois Light Company, Illinois Power Company, and Kansas City Power and Light Company, formed a joint study team with Babcock and Wilcox and presented a conceptual design of a single-fluid, 150,000 kw(e) reactor power plant developed by Babcock and Wilcox.[119] To avoid the slurry problems usually associated with homogeneous breeder reactors, a single-fluid, two-region reactor with a blanket consisting of assemblies of beds of thorium oxide pellets was selected. The assemblies would be periodically rotated to equalize the breeding of uranium-233 in the pellet beds.

In developing the Homogeneous Aqueous Reactor (HAR) concept, British workers studied many types before settling on the two-region power breeder, similar in several ways to the Thorium Breeder Reactor proposed by ORNL. More than one version of this two-region pressurized reactor have been described.[120-123]

A Japanese group considered an Aqueous Homogeneous Breeder Reactor similar to that of ORNL.[124] The fertile material would be thorium oxide slurry in heavy water so as to take advantage of Japan's abundant thorium supplies.

A Russian non-boiling two-region reactor has been reported, but few details have been given.[125] Plutonium is bred from a solution of a natural uranium salt in heavy water enriched with plutonium produced in the same reactor.

A concept for a Swedish two-region breeder is described in a French patent.[126] It is claimed to have the advantages of both one- and two-region breeders. Alternative fuels are a solution of uranyl sulfate, suspension of uranium oxide, or solution of plutonium sulfate, all in heavy water. The fertile material is either a slurry of thorium oxide, a slurry of uranium oxide, or a thorium nitrate solution, also in heavy water. Means are provided to exchange material between core and blanket and to separate slurry into solids and liquid.

Non-Homogeneous Aqueous Solution Reactors

Several reactor concepts that have been proposed include aqueous solutions in a structured core in the presence of a massive moderator or in a container.

Most of these will be included under "Semi-Homogeneous Reactors" in Part II. One concept that will be covered here is the reactor in which a uranium solution is contained in tubes. In a reactor designed for power and for breeding uranium-233, a solution of uranium-233 in heavy water is contained in tubes, around which a fertile slurry of thorium-232 in (preferably) heavy water flows. The slurry extracts heat from the fuel.[127]

Colloidal Fuels

Colloidal suspensions (sols) have been suggested as nuclear fuels. No complete reactor concept has been developed, but workers at Ionics, Inc., have proposed use of urania-thoria sols in water.[128] Particle size is approximately 25 mμ. They prepared such sols that were stable for several hundred hours at 250-300°C. Because of their neutral environment, very slow settling, and non-caking characteristics, such sols are claimed to ameliorate the conditions of corrosion and phase instability of solution fuels, and the settling, caking, and erosion difficulties of slurries. No work, however, appears to have been done on radiation stability or other properties that would be significant in a reactor.

Slurry-Fueled Reactors

In general, aqueous-slurry (suspension) reactors are nearly defined by their name. Many definitions have been given, but most differ only in details. In this chapter, the definition of Lane, McPherson, and Maslan[129] will be used. Aqueous slurry fuels are suspensions of loose and relatively independent clouds of joined particles (flocs) that are large enough to settle. The particle size varies widely, depending on the material, but 50 to 500 mμ covers them generally. They thus differ from colloids, in which the particles are so finely divided that their surface attraction forces exert a strong influence on the mechanical properties of the material as a whole. Normally colloidal particles do not settle out on standing.

Slurries attracted attention early as possible reactor fuels. In fact, as mentioned in Chapter 2, Von Halban and Kowarski used a suspension of uranium oxide in heavy water to demonstrate the possibility of a self-sustaining nuclear reaction. Their several advantages include the possibility of obtaining higher concentrations of fissionable material and freedom from the corrosivity of acid solutions. Their use, however, also has disadvantages, which include caking and settling of the solid particles, erosion, and pumping difficulties. Solving these problems has been a large part of the research and development on these reactors. As with solution reactors, no commercial reactors using slurries have been built, although several have been designed, and the feasibility of such reactors has not yet been conclusively established.

The early history of slurry reactors generally parallels that of solution reactors. There was early interest and extensive research, which later diminished until the renewal of interest with the HRE project. Lane, MacPherson, and Maslan have reviewed the early development.[4] Workers at the University of Chicago, Columbia University, the Clinton Laboratories and other laboratories studied the behavior of slurries, difficulties of using them, and reactor concepts that could be based on them. This research continued until near the end of 1944, when most of it ended. Some continued, however, especially at the Clinton Laboratories.

The early work resulted in some reactor concepts, and provided valuable information for later work under the Homogeneous Reactor Project. One of the earlier concepts was for a homogeneous production and power pile using a slurry of uranium oxide in heavy water.[130] The slurry is pumped through pipes to heat exchangers or flash evaporators, then back to the pile.

At Columbia, a general concept for a slurry reactor was reported in 1944.[131] A slurry of slightly enriched (uranium-233) uranium trioxide is suspended in heavy water and, presumably, cooled by circulation to external heat exchangers. The power was estimated at 600 Mw. Other concepts from the Manhattan Project included the Homogeneous Slurry Pile[132] and the Heterogeneous Slurry Pile.[133] Most of the other early slurry concepts either closely resemble these concepts or have been incorporated into later designs.

Slurry Development in the HRE Project

Before and during the Homogeneous Reactor Project, extensive development was carried out at Oak Ridge on slurries and slurry reactors. Such problems as the preparation of suitable slurries, prevention of caking and settling, and determination of optimum particle sizes were investigated, and designs for slurry reactors were formulated. Much of this work is reported in the Quarterly Progress Reports of the Homogeneous Reactor Project, particularly some of those for 1950-1952.[134-136]

A reactor originally proposed in 1943, the Natural-Uranium-Plutonium Producer, was considered by the Homogeneous Reactor Project in 1949.[90] In this one-region converter, the fuel is a slurry of natural uranium oxide in heavy water. The alternative fuel considered was a solution of uranyl fluoride in heavy water. For this fuel, the reactor would be operated at high pressure. This concept was considered to offer the possibility of being the most economical plutonium producer.

A slurry reactor was conceived as an alternative to the solution breeder for central-station power described under "Two-Region Converters and Breeders."[111] It is a single-region breeder, with a slurry of thorium dioxide and uranium trioxide in heavy water serving as fuel, coolant, moderator, and fertile material. The power is rated at 100 Mw(e).

A one-region uranium-238 converter and power reactor was one of the reactors evaluated in 1956 for possible large-scale development.[59] In this reactor, a slurry of uranium trioxide and plutonium dioxide in heavy water is the fuel. It is designed for a conversion ratio of 0.88 and production of 440 Mw(t), or 100 Mw(e) from the generating plant.

In 1956, nuclear characteristics were computed for more than 400 two-region spherical reactors having a slurry of thorium dioxide and uranium-233 dioxide in the core and blanket.[137]

After the solution-fueled reactors in the HRE sequency, HRE-4 was designed as a slurry-fueled reactor.[138,139] The purpose was to study such aspects as criticality, effects of radiation on slurries, and operational problems of a slurry-fueled reactor. It is a one-region reactor fueled with a slurry of thorium dioxide and enriched uranium dioxide in heavy water. The system is pressurized by boiling slurry in the dump tank. The original design power level was 5 Mw(t). It was planned to operate HRE-4 later with higher concentrations of slurry in the blanket, concentrations approaching those in the blanket of a two-region breeder. A later report[140] refers to HRE-4 as a 100-kw critical system, with a moderator of 80 percent heavy and 20 percent light water.

A reactor concept developed late in the Homogeneous Reactor Project was a power breeder with a slurry core and blanket.[60] The fuel is a slurry of thorium dioxide and enriched uranium oxide in heavy water. The fertile blanket is a slurry of thorium oxide in heavy water. This reactor was designed for high power--1140 Mw(t) or 333 Mw(e).

ORSORT Concepts

Students at the Oak Ridge School of Reactor Technology designed slurry reactors for generating electrical power. In the reactor for electrical power in a stationary plant, the "Homogeneous Power Producer,"[141] the core is a stainless-steel sphere, with concentric shells. It is cooled by water circulating between the inner and outer shells. The fuel-moderator-coolant-fertile material is a slurry of natural uranium in heavy water. The plutonium produced remains as supplementary fuel. In another slurry reactor for power, natural uranium trioxide, as a slurry in heavy water, is the fuel.[142] Two reactors are combined to produce a total power of 1000 Mw(t).

The PAR Reactor

The Dow Chemical Company and Detroit Edison worked on developing a slurry reactor for power.[143] It would be fueled with a slurry of uranium oxide or oxyfluoride in heavy or light water, and it would have a graphite core structure. It would be workable, but the need for expensive low-temperature, low-pressure steam installations resulted in the abandonment of this reactor.

The Pennsylvania Advanced Reactor (PAR) has been referred to as the second, and last attempt in the U. S. to use the concept of homogeneous aqueous fuels industrially.[144] The first was considered to be the Wolverine solution reactor. Thus the authors disregarded the Dow-Detroit Edison concept.

The PAR, also known as the Pennsylvania State Advanced Reactor, was a one-region aqueous slurry type using uranium oxide fuel and thorium oxide fertile material, with very fine particles--under 5 μ--suspended in heavy water, which acts as both moderator and coolant. The slurry is pumped through the spherical reactor vessel. It then goes to heat exchangers for steam generation. The power for a reference design is 550 thermal megawatts.[145]

The PAR Project originated in November 1954, when Pennsylvania Power and Light Company and Westinghouse Electric Corporation joined to survey feasibility of various reactor types for power generation. The PAR Project, formally set up in August 1955, was part of the Westinghouse commercial atomic power activities in Pittsburgh.[146] Shortly after the start of the Project, Union Carbide Nuclear Company joined the group to study the chemical-reprocessing problems involved.[147]

After two and a half years of the program, the AEC indicated willingness to support the research and development leading to the design of a plant;[148] in 1958, Pennsylvania Power and Light decided to abandon the project, but decided to develop a final reference design. A small prototype had been judged necessary before the final industrial design, but it was abandoned because it was considered too expensive.[149]

A major effort on the project was directed toward finding solutions to problems associated with transferring and circulating aqueous slurries of thorium oxide at the concentrations and conditions of a single-region homogeneous reactor.[150] The world's largest test loop for homogeneous reactors, capable of circulating 4000 gpm at the same temperatures and pressures planned for the power plant, was used in pumping slurry for 35,000 hours.

KEMA Reactors

The research laboratory of N. V. KEMA, the Netherlands, has instituted a program to develop a suspension-fueled power reactor. The three steps planned are: a subcritical assembly, a small-scale process reactor, and a power-demonstration reactor.[151] To date, only the first two steps have reached the concept stage.

The suspension-type reactor is known, together with its subcritical assemblies, as the Aqueous Homogeneous Suspension Reactor, Arnhem Suspension Reactor, BABYPOP, KSRT or KEMA Suspension Test Reactor, Subcritical Homogeneous Suspension Reactor, Suspension Test Reactor, SUSPOP, or 250 kw Homogeneous Suspension Reactor.

KEMA chose a suspension reactor, although it had been abandoned in the U. S., for several reasons.[152,153] The homogeneous reactor with fissile material distributed in heavy water needs no structured core and also has a high power extraction. Fission products can be removed from the reactor medium in a decontamination plant. No safety or control rods are required. Also a high conversion rate of fertile to fissile material with low fuel-fabrication and processing costs is possible.

The initial reactors were BABYPOP, and SUSPOP. BABYPOP was a zero-power reactor in which uranium dioxide or U_3O_8 is suspended in light water. Little has been published concerning it,[121] but it apparently has most of the features of SUSPOP, a subcritical assembly that would become critical later if possible. SUSPOP is fueled with an aqueous homogeneous suspension containing 20 percent enriched uranium as uranium dioxide.[151,153,154] It is moderated with light water and reflected by beryllium oxide and graphite. R. C. N. (Reactor Centrum Nederland) owns SUSPOP, which KEMA has operated since 1955 to investigate the flow behavior of a circulating suspension reactor.

The second stage, or process reactor, is also known as the KEMA Suspension Test Reactor. It has been designed as a one-region reactor fueled with a suspension of 20 percent enriched uranium dioxide suspended in light water.[154-156] It is pressurized with hydrogen to reduce oxidation, water decomposition, and stress corrosion. The design power is 250 kw. This reactor is intended for investigating, under power-reactor conditions, chemical and mechanical properties of the suspension, water decomposition, and special purification systems that had been suggested. The power density, temperatures, and suspension concentration are meant to equal those in a normal power reactor. This reactor was originally scheduled for construction after the zero-energy reactor had operated satisfactorily.

The third stage is intended to be a simplified power demonstration reactor in which important reactor components--vessels, valves, pumps, etc.--would be investigated, as would the possibility of remote maintenance of such an installation.

Other reactors were considered briefly.[157] A heterogeneous suspension reactor might use natural uranium if it were heterogeneous both geometrically and thermally. A very concentrated suspension would flow through tubes inside the reactor tank filled with heavy water. Just after leaving the reactor the concentrated suspension would be diluted slightly so it could easily flow through the external heat exchanger. A homogeneous reactor comparable to SUSPOP, was also considered. It would use light water (in dilute suspension only) as both moderator and suspension liquid for uranium trioxide or U_3O_8.

In 1962 it was reported that work in SUSPOP was continuing, in collaboration with teams at Harwell, Mol (Belgium), and Saclay (France).[158] Construction of the test reactor was under discussion.

Other Slurry Concepts

In addition to those concepts already discussed, other slurry-fueled reactors have been considered in England, Canada, Czechoslovakia, and Russia, with varying degrees of development taking place.

In developing the Homogeneous Aqueous Reactor (HAR), the British Atomic Energy Research Establishment studied slurries and considered them as possible fuels. In 1954, the two-region pressurized reactor, with a solution of uranium-233 salt in heavy water and a blanket of thorium oxide slurry, was chosen over the single-zone slurry reactor. The choice was made because the single-region slurry and other reactors either were less economic, required new developments, or had stability or safety problems.[159]

Later, however, when the system again came under review, the physics of a single-region slurry system were reassessed. The reactor is assumed to be an unreflected sphere of 7 to 15 feet in diameter, containing a slurry of thoria with uranium-233 in heavy water.[160]

The Atomic Energy Commission of Canada in 1953 made calculations on a two-region, heavy-water-slurry fuel reactor to assess the benefits of a moderate amount of plutonium enrichment.[161] They concluded that the matter deserved further study. In one concept, the reactor is fueled with a heavy-water slurry of natural uranium oxides enriched with plutonium, with a fertile slurry of natural uranium oxide as a blanket. The reactor consists of two concentric spheres, with the uranium oxide slurry between the spheres, and the inner sphere containing the plutonium.

HR-1 and HR-2 are Czechoslovak reactors that are slurry-fueled. HR-1 is a 10-Mw one-region homogeneous reactor, in which the fuel is a suspension of 15 percent enriched U_3O_8 in light water. HR-1 is an experimental pressurized homogeneous reactor suitable for research work and for obtaining practical manufacturing and operational experience in this field.[162] Experiments on the hydrodynamics of the suspension led to the HR-2 concept.[163] The basic concept of HR-2 is an arrangement in which the slurry moves only in the vertical plane to eliminate precipitation of the solid phase. The active zone, which is in the lower part of the reactor, consists of two coaxial components. The suspension flows downward in the outer part and upward in the inner. Work on this reactor was discontinued.[164]

Evaluation and Status

The extensive developmental programs on aqueous homogeneous reactors as power and breeder reactors have been followed by several evaluations of these breeders for industrial promise. Some evaluations of individual types have been discussed previously in this chapter.

One of the earlier evaluations of these reactors was that made by the General Electric Company in 1955. They found that these reactors must be considered as a promising approach to expansion and development of atomic power.[165]

In 1956, Foster Wheeler Corporation and the Pacific Northwest Power Group reported they were considering the HRT or TBR type; Foster-Wheeler proposed building a two-region homogeneous thorium breeder for delivery in 1961 or 1962; the exact type was unspecified.[34]

The program of the Homógeneous Reactor Project was one of those evaluated by the Fluid Fuel Reactors Task Force. In January 1959, the Task Force decided to reduce work on fluid fuels and consolidate it into one program at the Oak Ridge National Laboratory, with basic research to be carried out for an undetermined period.[166]

This decision was made despite the more-favorable report by members of a review and evaluation group at Argonne National Laboratory and the opinion of Lane. The Argonne group concluded that aqueous homogeneous reactors offer many advantages as plutonium producers, but they must be considered as long-range possibilities because of present difficulties.[167]

In March 1959, Lane commented:

> .."the consensus of those associated with the Homogeneous Reactor Program is that, in spite of technical problems which have been encountered since the inception of the program, the potential of aqueous homogeneous systems is as great as it ever has been. This conclusion is based on the most recent information."[168]

In April 1961, HRT was permanently shut down and the fuel was processed for recovery of uranium-235. Work on the aqueous homogeneous reactor concept was terminated on June 30, 1961, although some research on aqueous systems was continued under the Thorium Utilization Program, initiated at Oak Ridge during the latter half of 1961.[25] The thorium program is related to the previous program to develop a thorium-uranium-233 breeder.[169]

Another evaluation of thorium breeder reactors, made in 1961, concluded that the Aqueous Homogeneous Breeder Reactor ranked first over the molten-salt, liquid-bismuth, and gas-cooled types in regard to nuclear capability, fuel-cycle potential, and status of development.[62]

To date, the current status of developmental programs and the actions by the Atomic Energy Commission and by industry indicate that the future commercial development of aqueous homogeneous reactors--either solution or slurry--remains uncertain.

D A T A   S H E E T S

NON-BOILING REACTORS FOR POWER AND BREEDING

No. 1 Pilot Model for HRE Series

ORNL

Reference: CF-49-7-135 Del.

Originator: A. M. Weinberg *et al*.

Status: Conceptual design, July 1949; predecessor of HRE series.

Details: Thermal neutrons, steady state, single-region converter. Fuel-moderator: solution of enriched uranyl sulfate in H_2O: 2 kg UO_2SO_4 per 50 liters; 40 g U^{235} per liter. Moderator-reflector: D_2O in 15-cm blanket around reactor. Fuel solution, at 482°F and 1000 psi, is pumped downward from the reactor through two vertical heat exchangers and cooled to 437°F by heat exchange with boiling water at 400°F and 250 psig. Steam is produced at 600 lb per hour. The D_2O moderator is cooled by coils immersed in retention tank for D_2O. Reactor vessel: cylindrical tank, 40 cm diameter, 40 cm high. 50 liters volume. Controls: fine control by vertical regulating rods in enclosed thimbles around reactor vessel; raising and lowering level of D_2O blanket is control (without blanket, reactor is non-critical). For rapid shutdown, blanket and fuel can be emptied into dump tanks. Power: 20 kw(e).

Code: 0311 14 31201 43 624 744 8111X 921 101
82188
83189

No. 2 HRE-1, Homogeneous Reactor Experiment

ORNL

References: ORNL-527; TID-10082; ORNL-730; *J. Brit. Nuc. Eng. Soc.*, 1, p. 35, January 1962; *IAEA Directory of Nuclear Reactors*, 2, p. 153; Lane, et al, *Fluid Fuel Reactors*, pp. 348-59.

Originator: ORNL staff.

Status: Critical April 1952; operating at full power spring of 1954. Dismantled.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: solution of 93.4% enriched U^{235} as uranyl sulfate in light water circulating through closed circuit, spiral path in core, at 100 gpm. Coolant: circulating fuel. Other cooling: water-spray on reflector pressure shell, natural convection water system on solution and reflector dump tanks. Operating temperature in core: 250°C; pressure of liquids in both core and reflector: 1000 psi. Reflector: D_2O pressurized by helium, contained in 39 in. ID forged steel pressure vessel surrounding core, and cooled by recirculation through a heat exchanger. Stainless-steel spherical core, 18 in. ID, 50 liters capacity. Control: changing concentration of solution: 2 shim and safety rods and 1 regulating rod (boral plates clad with stainless-steel and bent to form segments of vertical cylinder around core) in reflector zone tangent to core; and changing reflector level. MTR-type operating mechanism. Pilot-plant power level: 200-1000 kw. HRE-1 demonstrated the stability of a circulating fuel reactor operating at high power densities. It was planned to substitute certain parts, i.e., reactor vessel of Zr instead of stainless steel, later in the experiment. Consequently, unmortared concrete blocks were used for shielding. The substitutions later were incorporated in HRE-2.

Code:								
0311	13	31201	44	624	711	83779	921	101
						81141		
						82188		

No. 3 HRE-2, Homogeneous Reactor Experiment-2

(HRT, Homogeneous Reactor Test)(HRE-2/HRT)

ORNL

References: ORNL-2096; ORNL-2148; ORNL-2222; ORNL-2272; ORNL-2379; ORNL-3167; Proc. 2nd U.N. Int. Conf., 9, pp. 509-527; Lane, et al, Fluid Fuel Reactors, pp. 359-97; Proc. 1st U.N. Int. Conf., 3, pp. 263-82.

Originator: ORNL staff.

Status: Critical December 1957; permanently shut down, 1961.

Details: Occupied same building that housed HRE-1. Thermal neutrons, steady state, burner. Fuel-moderator: 93% enriched U^{235} as UO_2SO_4-$CuSO_4$-H_2SO_4 in D_2O. $CuSO_4$ added to supress radiolytic decomposition of water. Coolant: circulating fuel solution. Reflector: D_2O, 13.69 in. thick between core tank and stainless-steel clad carbon steel pressure (2000 psi) vessel (60 in. ID); cooled by circulation through heat exchanger. Average temperature for both fuel solution and moderator: 280°C; pressure: 1700 psi. Core tank: inverted-pear shape (top section 32 in. ID, spherical) of Zircaloy-2, 290 liters volume. Designed as two-region reactor, but hole in core tank let fuel into reflector. Consequently run as one-region. Control: no control rods; ratio of heat withdrawal from reactor (power demand at turbine) automatically met by negative temperature coefficient ($\sim$0.3% Δk/°C at 280°C). Normal control by dilution of fuel with D_2O. Power in core: 5 Mw(t). HRE-2 was built to answer questions of reliability, performance, and maintenance in long-term operation. Construction completed in April 1956. From then until December 1957, non-nuclear tests were performed.

Code: 0311 14 31202 44 625 711 84677 921 101
83779

No. 4 HRE-2 Replacement Reactor

ORNL

Reference: CF-61-7-54; CF-59-8-110.

Originator: R. H. Chapman.

Status: Reference design report as a result of termination by AEC of HRP.

Details: Thermal neutrons, steady state, breeder (when blanket is used as described here). Fuel-moderator-coolant: solution of U^{235} (presumably highly enriched uranyl sulfate) in D_2O. Reflector: 4-in. thick Be surrounding Zircaloy-2 cylindrical core tank (21 in. ID x 42 in. long). Fuel enters tangentially through two "slotted-entry" headers on the cylindrical portion of the tank with sufficient velocity to provide proper cooling of the wall. At each end of the headers, the slots direct some flow into the hemispherical ends; the major portion leaves through an outlet on the top of the tank. An 8-in.-thick ThO_2 pellet blanket surrounds the Be. Fuel and blanket temperature: 280°C. Breeding ratio: estimated at 1.07 for 6 in. blanket. Control: (probably) by negative temperature coefficient. Power: 5-10 Mw. This design is for an experimental reactor suitable for installation in the existing HRE-2 facility with a minimum of system changes.

Code: 0312 14 31202 44 625 786 84677 941 101

No. 5 Intermediate-Scale Homogeneous Reactor (ISHR), Single Region Design

ORNL

References: ORNL-1280, pp. 113-33; ORNL-1318, pp. 95-111; ORNL-1424, pp. 43-53; Lane, et al, Fluid Fuel Reactors, pp. 8, 504-5; CF-52-8-31.

Originators: Homogeneous Reactor Project staff.

Status: Design studies commenced October 1952; further work deferred late in 1953 because "construction...[of HRE-2]...seemed a more suitable course of action." HRP cancelled 1961.

Details: Thermal neutrons, steady state, converter. Fuel-moderator: 2.5% enriched U^{235} as uranyl sulfate in D_2O. Solution temperature: 250°C. Both vortex and axial flow studied as possibilities. Core vessel: spherical stainless steel, 6 ft in diameter. Maximum pressure: 1000 psia. No reflector other than the steel core shell. Cooling water circulates through three 3/4-in.-wide passages between three 3/4-in. thick steel shells, all of which are enclosed by a 2-1/2-in.-thick steel outer shell. Main circulating system: reactor vessel, external contrifugal gas separator, shell-and-tube heat exchanger (vertical, forced-circulation boiler), and circulating pump. Control: assumed through varying concentration of solution and negative temperature coefficient. Reactor designed as a power and plutonium producer. Power: 48 Mw.

Code: 0311 14 31202 42 625 743 84677 921 101
83779

No. 6 Cooling Tower Reactor

ORNL

Reference: CF-53-10-185.

Originators: J. P. Sanders and P. N. Haubenreich.

Status: Preliminary design, 1953; no further work.

Details: Thermal neutrons, steady state, burner, converter, or breeder. Based on the ISHR Single-Region design, the Cooling Tower Reactor differs mainly in the method of removing heat from the solution. Fuel-moderator: UO_2SO_4 solution in D_2O at 700 psia. Solution enters the packed or slotted cooling tower at 482°F (250°C as in ISHR) and leaves at 416°F. A portion of the D_2O is evaporated into a recirculating gas stream, which is then cooled. Part of the vapor is condensed in a heat exchanger producing steam. The tower bed and the core are located in a horizontal pressure vessel 10 ft in diameter and 30 ft long. The reactor may be used as a breeder, or, by substituting H_2O for D_2O, as a burner to produce power.

Code:	0311	14	31202	42	625	743	84677	921	101
	0312	13	31201	44	624	711			
	0313								

No. 7 Intermediate Scale Homogeneous Reactor (ISHR) Two-Region Converter

ORNL

References: ORNL-1424, pp. 53-7; ORNL-1478, pp. 23-40; ORNL-1605, pp. 31-46.

Originators: HRP staff.

Status: Design studies, 1952-53; emphasis shifted in October 1952 from one- to two-region design. ISHR deferred in fall 1953 for work on HRE-2. HRP cancelled, 1961.

Details: Thermal neutrons, steady state, converter with blanket conversion. Fuel-moderator: 93.5% enriched U^{235} in dilute D_2O solution of uranyl sulfate. Fertile material: breeding blanket, 2 ft thick, of ThO_2-D_2O slurry. Other blanket compositions considered were: thorium for a breeder, lithium compound to produce tritium, and uranium to produce plutonium. Both solution and slurry are under 1000 psia pressure at 250°C. A modification of the ISHR Single-Region design, the core vessel of the ISHR Two-Region Converter is a sphere, 4 ft diameter, of stainless steel. Fuel enters at the bottom, where it is heated to 482°F, and leaves at the top to return to the two-pass, horizontal, shell-and-tube heat exchanger for cooling. An arrangement of screens in the core tank and the small-diameter holes through the pressure shell from the breeder material inlet and outlet headers provide efficient flow of fluids through the vessel. Cooling by thermal convection in the high-pressure recirculating system makes fuel dumping unnecessary. Control: control rod designs were considered but discarded on basis of self-regulation by temperature coefficient. Power: core-48 Mw; blanket-9.6 Mw.

Code: 0311 14 31202 44 625 756 84677 941 101

No. 8 One-Region Breeder ISHR Design

ORNL

Reference: ORNL-1605, pp. 46-7.

Originator: HRP staff.

Status: Calculations completed, 1953; discarded.

Details: Thermal neutrons, steady state, breeder. Fuel: U^{233}. Moderator-coolant: D_2O. Fertile material: Th^{232}. They are mixed together in single core 10, 15, or 20 ft in diameter. No reflector. Constants are the same as for two-region reactors. Power: between 500-2000 Mw.

Code: 0312 14 31X02 45 6X5 7X6 84677 91X 101

No. 9 Two-Region Breeder, ISHR Design

ORNL

Reference: ORNL-1605, pp. 44-5.

Originator: HRP staff.

Status: Calculations completed for various core and blanket sizes, 1953; discarded.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator: same as for the ISHR Two-Region Converter. Fertile material: blanket containing ThO_2 pellets or slurry. Core diameter: 6 ft. Slurry blanket would be 2 ft thick and the pellet blanket 1 ft thick, although the latter may have to be larger. Other details the same as for the converter. Power (core): 320 Mw.

Code: 0312 14 31202 44 625 756 84677 941 101
786

No. 10 Intermediate Scale Lithium Converter Three-Region Reactor for Production of Tritium

ORNL

Reference: ORNL-1478, p. 40-1.

Originator: HRP staff.

Status: Conversion calculations completed, 1953; discarded.

Details: Same as ISHR Two-Region Converter Design except that core tank is surrounded by a 3-cm layer of molten Li as a blanket which, in turn, is surrounded by a BeO reflector.

Code: 0311 14 31202 44 625 738 84677 941 101

No. 11 Two-Region Thorium Breeder Reactor (TBR) (K-23 Reactor)

ORNL

References: TID-7524, pp. 65-77; Lane et al, Fluid Fuel Reactors, pp. 507-9; Proc. 1st U.N. Int. Conf., 3, pp. 175-187; ORNL-1761.

Originator: HRP staff.

Status: Conceptual design completed, 1955 (unpublished); set aside because of urgent work needed for HRT; no further work.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator: solution of UO_2SO_4 (mainly U^{233}, with U^{234}, U^{235}, and U^{236}) in D_2O. Fuel flows into core through center pipe (at 250°C) and leaves through the annulus at the top (at 300°C). Fertile material: blanket, or breeding region (27 in. thick) containing slurry of ThO_2 in D_2O, which enters at the bottom of blanket vessel and exits at the top. Breeding ratio is 1.11 (gross). Pressure (core solution and blanket slurry): 2000 psi. Core vessel: sphere, 5 ft diameter (6-ft-diameter core was first considered), of Zircaloy-2, and blanket pressure vessel, 10.5 ft diameter, of stainless-steel-clad carbon steel. Vertical steel envelope encloses all parts of reactor plant that handle radioactive fluids under pressure. H_2O flows through heat exchangers built into core and blanket circulating pumps to remove excess heat. Main heat exchangers: core and blanket steam generators. Control: high negative temperature coefficient. Three reactors comprise proposed plant; each produces 100 Mw(e)--440 Mw(t)--for plant total of 300 Mw(e) net.

Code: 0312 14 31202 45 625 756 84677 941 101

No. 12 HRE-3 Homogeneous Reactor Experiment-3

ORNL

References: CF-58-11-112; Lane, et al, Fluid Fuel Reactors, pp. 509-11.

Originator: HRP staff.

Status: Design of HRE-3 as a two-region power breeder reactor was suspended pending resolution of problems arising in operation of the HRT. Preliminary design was summarized, November 1958. Design studies discontinued, 1958, to investigate slurry-handling problems. Homogeneous Reactor Project cancelled, 1961.

Details: Thermal neutrons, steady state, two-region, thorium breeder. Fuel-moderator-coolant: U^{233} as uranyl sulfate solution in D_2O. Fertile material: 2-ft blanket of slurry, ThO_2 in D_2O. Pressure of core and blanket: 1500 psi; average temperature of core: 280°C. Oxygen is favored as pressurizing medium for core and blanket. Core tank: Zr sphere, 4-ft diameter inside pressure vessel, 9-ft diameter, although a cylindrical vessel 3-ft diameter and 6-ft high considered. Sphere volume: about 950 liters. Pressure vessel: carbon steel clad with stainless-steel. Control: not described, assumed to be self-regulation by reactivity temperature coefficient. Power: 50 Mw(t) in core; 10 Mw(t) in blanket.

Code: 0312 14 31202 45 625 756 84677 94 101

No. 13 U^{235} Burner

ORNL

Reference: CF-56-1-26, Del., p. 9.

Originator: HRP staff.

Status: One of five reactors evaluated by HRP for possible large-scale development in January 1956; considered most feasible. Project terminated 1961.

Details: Thermal neutrons, steady state, one-region burner. Fuel-moderator-coolant: uranyl sulfate solution in D_2O, with $CuSO_4$ added; 1.8 g uranium per liter; 0.54 g fissile material per liter. Maximum fuel temperature: 300°C. Reactor pressure: 1800 psia. Total system volume: 32,000 liters. Reactor diameter: 10 ft. Power: 440 Mw(t); 100 Mw(e) from generating plant.

Code: 0313 14 31202 43 625 711 84677 91 101
8XXXX

No. 14 One-Region U^{238}-Pu Converter and Power Reactor

ORNL

Reference: CF-56-1-26, Del., p. 9.

Originator: HRP staff.

Status: One of five reactors evaluated for possible large-scale development, January 1956. Project terminated 1961.

Details: Thermal neutrons, steady state, one-region converter. Fuel-moderator-coolant: uranyl sulfate solution in D_2O with $CuSO_4$ added; 300 g uranium per liter; 4.1 g fissile material per liter. Maximum fuel temperature: 250°C. Reactor pressure: 1000 psia. Total system volume: 46,000 liters. Reactor diameter: 12 ft. Conversion ratio: 0.81. Power: 480 Mw(t); 100 Mw(e) from generating plant.

Code: 0311 14 31202 42 625 743 84677 91X 101

No. 15 One-Region Th-U^{233} Breeder

ORNL

Reference: CF-56-1-26, Del., p. 9.

Originator: HRP staff.

Status: One of five reactors evaluated for possible large-scale development, January 1956. Project terminated, 1961.

Details: Thermal neutrons, steady state, one-region breeder. Fuel-moderator-fertile material: slurry of ThO_2 and UO_3 in D_2O; 300 g uranium per liter; 6.2 g fissile material per liter. Maximum fuel temperature: 300°C. Reactor pressure: 1800 psia. Total system volume: 46,000 liters. Reactor diameter: 12 ft. Conversion ratio: 1.0+. Power: 440 Mw(t); 100 Mw(e) from generating plant.

Code: 0312 14 31302 45 635 753 84677 91X 101
756

No. 16 Two-Region Th-U^{233} Breeder

ORNL

Reference: CF-56-1-26, Del., p. 9.

Originator: HRP staff.

Status: One of five reactors evaluated for possible large-scale development, January 1956. Project terminated, 1961.

Details: Thermal neutrons, steady state, two-region breeder. Fuel-moderator-coolant: uranyl sulfate solution in D_2O, with $CuSO_4$ in core; 6.3 g uranium per liter; 2.4 g fissile material per liter in core, plus 3 g per liter in blanket. Fertile material: slurry blanket of ThO_2 and UO_3 in D_2O; 1000 g per liter. Maximum fuel temperature: 300°C. Reactor pressure: 1800 psia. Total system volume: 27,900 liters. Reactor diameter: 9 ft, with 5 ft diameter core. Conversion ratio: 1.1. Power: 440 Mw(t); 100 Mw(e) from generating plant.

Code: 0312 14 31202 45 625 754 84677 941 101
756

No. 17 400-Mw Thorium Breeder with Pellet Blanket

ORNL

Reference: Unpublished report, ORNL, Jan. 4, 1960.

Originators: I. Spiewak and F. N. Peebles.

Status: Conceptual design for evaluation, Jan. 4, 1960; project terminated, 1961.

Details: Thermal neutrons, steady state, two-region breeder. Fuel-moderator: UO_2SO_4 solution in D_2O; 5 g total U per liter; 1.55 g U^{233} plus U^{235} per liter. Fertile material: 2 ft blanket of 1/8-in. thorium oxide pellets, 1450 g Th per liter. Fuel solution enters at top of core, at 240°C and 26,600 gpm, through swirl-producing vanes. Fluid blankets the whole inner surface of the reactor tank. Fuel leaves at 290°C through top of reactor to go to external heat exchangers. Th pellets are in three annular rings, which are cooled by flowing D_2O. Core pressure is 20 psi above blanket pressure. Core vessel: Zircaloy-2 cylinder, 3/8 in. wall thickness, 4 ft diameter, 12 ft long. Control: negative temperature coefficient of solution and emergency dumping. Power: 464 Mw(t), 135 Mw(e). Similar design for 40 Mw test reactor.

Code: 0312 14 31202 45 625 786 84677 941 101
83189

No. 18 Solution-Core, Slurry-Blanket Power Breeder

ORNL

Reference: CF-59-7-129, pp. 11-18.

Originators: HRP staff.

Status: Conceptual design for a large breeder plant, 1959; HRP cancelled, 1961.

Details: Thermal neutrons, steady state, two-region breeder. Fuel-moderator: enriched uranyl sulfate solution in D_2O, 5 g uranium per liter (1.9 g U^{233}, 1.4 g U^{234}, 0.2 g U^{235}, 1.4 g U^{236}). $CuSO_4$, $NiSO_4$, and D_2SO_4 added. Fertile material: 2 ft thick blanket of ThO_2 in D_2O slurry, with U^{233} and Pa^{233}. Slurry: 1000 g Th per liter (11 vol. % solids); 2.5-4.5 g U^{233} plus Pa^{233} per kg thorium. Fuel solution enters core tank at 250°C and 33,000 gpm. It leaves at 290°C and divides into two parallel circuits, goes to steam generators, and treatment equipment, and returns to core. Core pressurized by boiling D_2O in surge chamber. Slurry circulates at 15,000 gpm, enters blanket at 240°C, and leaves at 250°C. Pumped to heat exchangers to heat feedwater for generators. Reactor vessel: zirconium-alloy cylinder, 4 ft diameter, 12 ft long. Control: regulating release of steam to generator and negative temperature coefficient. Power: 380 Mw(t). Three reactors combined to furnish steam to one 333 Mw(e) generator.

Code: 0312 14 31202 47 625 756 84677 941 101
31302

No. 19 Two-Region Homogeneous Solution Breeder

ORNL

Reference: ORNL-2920, pp. 27-36.

Originators: HRP staff.

Status: Preliminary mechanical design, January 1960; project cancelled, 1961.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator: designated as a solution but composition not given. Fertile material: slurry blanket; composition not given. Core: annular-inlet, polar-outlet type with swirling flow. Incoming fuel is evenly distributed in the annular inlet by a double-volute distributor. Blanket, a modified polar-inlet, annular-outlet type, provided with a single-volute discharge collector to distribute the flow evenly in the outlet. Core vessel: a 4 ft x 12 ft cylinder; core walls cooled to keep the inner surface below ∿ 260°C by circulating a cooled stream of D_2O through the annular gap between double core walls. A 6 in. shroud surrounding the core provides passage for upflowing slurry. Total blanket width: 2 ft. Coolant flow: 300 gpm; inlet temperature: 183°C; coolant outlet temperature: the same as inlet fuel temperature, 250°C. Power: 380 Mw(t).

Code: 0312 14 31202 4X 62X 75X 84677 941 101

No. 20 Aqueous Homogeneous Breeder Reactor (AHBR)

ORNL

Reference: CF-61-3-9, pp. 51-67a.

Originators: L. G. Alexander, W. L. Carter, R. H. Chapman, B. W. Kinyon, J. W. Miller, and R. Van Winkle.

Status: Typical concept used for evaluating purposes, 1961. Authors judged AABR first in regard to nuclear capability, fuel cycle potential, and status of development; however, homogeneous program cancelled by AEC in 1961.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator-coolant: solution of UO_2SO_4 in D_2O (no enrichment or isotope given). Fertile material: blanket 2 ft thick of 3 equal-volume annuli of ThO_2 pellets and 4 annuli of D_2O. Fuel solution enters core at 482°F through annular nozzle fitted with vanes to give it a swirling motion, which stabilizes the flow and prevents boundary-layer separation as the solution flows down along the core-vessel wall. Fuel returns along the axis of the core and exits at 554°F through a central nozzle leading to the pump, which discharges into two heat exchangers in parallel. Fuel returns to the core. Core vessel: 4.4 ft diameter x 14.6 ft long vertical cylinder of Zircaloy plate strengthened by ribs and stiffeners. Blanket pellets retained between Zircaloy-2 cylinders, concentric with the core vessel and perforated by small holes to a porosity of 50%. D_2O coolant flows up from the bottom of the vessel, along the core vessel wall, and then "percolates" through the 3 layers of pellets. Each layer is divided into 33 sections ("baskets") by Zircaloy plates. Blanket D_2O flows through pumps and then heat exchangers, from which it returns to the blanket inlet. Maximum allowable velocity for both fluids: 20 fps. Operating pressure: 2000 psi. Reflector: D_2O, 5.5 cm thick, surrounding blanket and core. Control: assumed to be by negative temperature coefficient. A station of four reactors would produce 3640 Mw(t) or 1000 Mw(e)--910 Mw(t) each.

Code: 0312 14 31202 4X 625 786 84677 941 101

No. 21 LAPRE-I, Los Alamos Power Reactor Experiment I

LASL

References: LAMS-1611; Lane, *et al*, *Fluid Fuel Reactors*, pp. 397-405; *Proc. 1st U. N. Int. Conf.*, *3*, pp. 283-6; *IAEA Directory of Nuclear Reactors*, *2*, pp. 159-64.

Originators: Originally designated DIR-P, project under Director's Office made up of two groups headed by R. P. Hammond and L. D. P. King.

Status: Critical February 1956; dismantled, January 1957.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: solution of UO_3 (93.4% enriched in U^{235}) in H_3PO_4 (acid concentration 50% by weight in H_2O). Coolant: H_2O pumped through coiled, gold-clad, stainless-steel tubing. Reflector: 3 in. thick steel wall of vessel and 4 ft water shield surrounding reactor. Fuel solution temperature: 430°C; pressure 3600 psi. Critical zone of core: cylinder 15 in. diameter and 16 in. high of gold-clad stainless steel. Core: test tube shaped; hollow boron cylinder divides non-critical lower portion of vessel into inner and outer section. Lower portion a storage for excess fuel, upper the critical region. Heat exchanger: between critical zone and storage region. Blanket gas: oxygen. Control: 1 shim safety rod and 4 regulating rods of platinum-clad stainless-steel thimbles filled with boron disks; vertical movement. Shim rod extends to bottom of vessel, regulating rods extend through reactor core region only. Power: 2 Mw(t). Reactor housed in cell at LASL which had been built to handle highly radioactive materials.

Code: 0313 13 31201 44 624 711 81111 921 105
110

No. 22 LAPRE-II, Los Alamos Power Reactor Experiment II

LASL

References: LA-2465; Lane, et al, Fluid Fuel Reactors, pp. 398-405.

Originators: Evidently same group that worked on LAPRE-I.

Status: Critical experiment scheduled February 1959; dismantled 1959.

Details: Simplified convection version of LAPRE-I. Thermal neutrons, steady state, burner. Fuel-moderator: 95% enriched $U^{235}O_2$ dissolved in 95% phosphoric acid-H_2O. Coolant: H_2O circulated through coils above reactor core. Core: cylinder 23.6 in. high and 14.75 in. ID. Core and coils constructed same as those of LAPRE-I. Solution circulates by natural convection over coils. Solution temperature: 800°F; pressure: about 800 psi. Hydrogen over-pressure: 200 psi. Reflector: 2 graphite concentric cylinders (sleeves). Shim control: BeO and graphite sleeve; no other controls. Power: 0.8-1 Mw(t). Reactor located in underground steel tank, 20 ft deep, 42 in. diameter.

Code:	0313	13	31201	44	624	711	82148	921	105
		110					84677		

No. 23 LAPRE-II Type Reactor for DEW Line Application

LASL

Reference: Unpublished internal report, LASL.

Originators: L. D. P. King, R. P. Hammond, G. I. Bell, P. J. Bendt, R. A. Clark, B. J. Melton, R. E. Peterson, and E. O. Swickard.

Status: Proposal, June 1955.

Details: Thermal neutrons, steady state, burner. Fuel-moderator-coolant: homogeneous solution of 93.5% enriched UO_2 in phosphoric acid with some H_2O; operated with 200 psi hydrogen overpressure. Secondary coolant: H_2O. Core: sphere 24 in. high, of stainless steel, silver, and gold. Reflector: graphite, 11 in. thick, in the form of a vertically-movable sleeve for shim control; is thicker than that for LAPRE-II. Other modifications of LAPRE-II include increased power (to 0.9-1.34 Mw) and insulated graphite sleeve with 6 in. foam glass. No control rods.

Code:	0313	13	31201	44	624	711	82148	921	101
							84677		

No. 24 Homogeneous Phosphoric Acid Reactor (HPAR)

SANDIA Corp.

Reference: SC-4459 (RR).

Originators: Sandia Corp., Div. 5433 staff.

Status: Design study, August 1960.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: solution of fully enriched UO_2 in H_3PO_4. Reactor vessel: gold-lined stainless-steel cylindrical pressure vessel, divided by cylindrical graphite flow-divider into reactor core portion (central) and heat exchanger portion (outer annulus). Pressure vessel: 20.25 in. OD (18 in. ID) and 45 in. high, contains the core, fuel circulator, and heat exchanger. Maximum pressure: 800 psia; fuel velocity: 20 ft/sec. Temperature at heat exchange inlet: 800°F; outlet: 795°F. Fuel is circulated by pump at top of vessel; graphite impeller and scroll are top parts of flow divider. Fuel solution enters at bottom of core, rises upward through core and is forced by pump down through outer heat-exchanger annulus. Secondary demineralized water-steam circuit of heat exchanger consists of spiral-wound heat-transfer tubes embedded (brazed behind gold) in walls of pressure-vessel liner. A separate superheat circuit is provided (top eight tubes). Forty tubes comprise boiler portion of heat exchanger. Reflector: shim reflector of BeO (3 in. thick) and graphite (3 in. thick) completely surrounding vessel in the form of two movable concentric cylinders; stationary graphite reflector outside the movable one. Control: mainly through negative temperature coefficient; shim reflector provides some control. Power level: 100-1500 kw(e). The reactor, designed for remote locations or mobile units, weighs 4200 lbs including reflector.

Code: 0313 13 31201 44 624 711 84677 921 101
82148

No. 25 500 Mw Mobile Reactor Design No. 1, 93.4% Enriched, H_2O-Moderated

ORNL

Reference: CF-50-10-114 Rev.

Originators: ORNL staff.

Status: Conceptual design, 1950. No further work.

Details: Thermal neutrons, steady state, burner. Fuel-moderator-coolant: solution of 93.4% enriched U^{235} as uranyl sulfate in H_2O. Core: 6 ft diameter stainless-steel lined, carbon-steel sphere. There are 6 horizontal shell-and-tube heat exchangers. Fuel solution is circulated by canned rotor centrifugal pumps at 25,000 gpm and is introduced through tangential inlets to the reactor core. The circular motion of the liquid in the core drives the gas bubbles toward the vortex and reduces the volume of vapor holdup in the core. A recombiner burns the gas to provide heat for superheating the steam. Solution inlet temperature: 208°C; outlet: 250°C. Reactor pressure: 2000 psia. Control: reactor is expected to be self-stabilizing; consequently, no control rods included. Power: dual units considered, each producing 250 Mw(t).

Code: 0313 13 31201 44 624 711 84677 921 101

No. 26 500 Mw Mobile Reactor Design No. 2, 3.5% Enriched, H_2O-Moderated

ORNL

Reference: CF-50-10-114 Rev.

Originators: ORNL staff.

Status: Conceptual design, 1950. No further work.

Details: Same general operating characteristics as Design No. 1. All other details are the same except for enrichment, which is 3.5% U^{235}, and the corresponding alterations in concentration of fuel, thermal flux, etc.

Code: 0311 13 31201 42 624 84677 921 101

No. 27 500 Mw Mobile Reactor Design No. 3, 1.5% Enriched, D_2O-Moderated

ORNL

Reference: CF-50-10-114 Rev.

Originators: ORNL staff.

Status: Conceptual design, 1950. No further work.

Details: Same operating characteristics as Design No. 1 with the following changes: fuel enrichment: 1.5%; moderator: D_2O; core: 9 ft diameter sphere. There are corresponding changes in fuel concentration, thermal flux, power density, etc. This design is considered by the authors as the most promising from the standpoint of initial fuel inventory and low fuel consumption.

Code: 0311 14 31202 42 625 743 84677 921 101

No. 28 Atomic-Powered Locomotive

University of Utah

References: Report, AIF Meeting March 15-16, 1954, New York, pp. c-2 to c-15; Nucleonics, 12, No. 3, March 1954, pp. 78-80.

Originator: L. B. Borst, with graduate students and fellow staff members.

Status: Proposal, 1954; no further work.

Details: Water-boiler type. Thermal neutrons, steady state, burner. Fuel-moderator: solution of fully enriched uranyl sulfate in H_2O. Coolant: H_2O circulated to remove heat. Reflector: 6 in. thick water. Core: 2 hexagonal slabs 3 ft square and 1 ft thick of stainless steel. Adjacent to each slab is a reflector, both connected by 10,000-kw heat-transfer tubes that penetrate core vessel. Core temperature: 455°F; reflector: 405°F. Control: 6 control rods that enter the core at a 45° angle; no other data given. Power: 30,000 kw(t) as steam at 250 psi to operate 7200-hp locomotive.

Code: 0313 13 31101 44 624 711 8111X 921 105

No. 29 Homogeneous Reactor for Ship Propulsion

ORSORT

Reference: CF-54-8-236 Del.

Originators: P. R. Clark, W. O. Chatfield, A. E. Cox, J. M. Detwyler, J. A. Murphy, C. W. Nestor, R. R. Roof, and D. H. Walker.

Status: Conceptual design and feasibility study; term paper, August 1954.

Details: Thermal neutrons, steady state, burner. Fuel-moderator-coolant: highly enriched uranyl sulfate in D_2O. Core: 5 ft diameter sphere; with concentric inlet and outlet; design chosen because its flow pattern is stable under rolling shipboard conditions. Operating temperature: 300°C; pressure: 2000 psi. All parts in contact with solution are stainless steel. Thermal shield cooled by D_2O flowing in 1 in. annulus between shield and pressure vessel. Complete internal recombination; no control rods or fission-product removal. Power: 75 Mw--10% more power than needed by the cargo ship, C-4 Mariner Class, for which it was designed.

Code: 0313 14 31202 44 625 84677 921 101

No. 30 Aqueous Homogeneous Circulating Solution Reactor

ORSORT

Reference: CF-53-10-22.

Originators: D. W. Montgomery, W. J. Dodson, F. F. Kaiser, W. K. Luckow, and T. J. Pashos.

Status: Conceptual design; term paper, August 1953.

Details: Thermal neutrons, steady state, burner. Fuel-moderator-coolant: circulating solution in D_2O of uranyl sulfate, approximately 93.5% enrichment of U^{235}. Pressure: 1000 psia; temperature: 250°C. Core: 4 ft diameter sphere. Pressure vessel: cylinder. Control: regulation of steam pressure, hence core temperature. Power: 10 Mw. "Package" power unit is removable, 17 ft 6 in. high and 6 ft 8 in. diameter.

Code: 0313 14 31202 44 625 711 84677 921 101

No. 31 STUPO, Homogeneous, Non-Boiling, Natural Convection Power Reactor

ORSORT

Reference: CF-53-8-225.

Originators: W. E. Kinney, R. Brodsky, D. Hillis, J. T. Wagner, and T. J. Ward.

Status: Conceptual design; term paper, August 1953.

Details: Thermal neutrons, steady state burner. Name is "extrapolation" of LOPO, HYPO, SUPO series as a result of an abortive attempt to extrapolate Los Alamos data. Fuel-moderator: 93.4% enriched U^{235} in D_2O. Zr baffle in center of core. Core diameter: 4-1/2 ft. Stainless-steel pressure vessel (8 ft 4 in. OD) provides some reflection. Solution heats in active core, rises through central pipe, enters upper sphere, passes through heat-exchanger tubes, and returns to core. Operating pressure: 1000 psi; temperature: 482°F. Natural convection circulation. Control: self-controlling; negative temperature coefficient expected to maintain level of operation. Power: 10 Mw max., 7 Mw average.

Code: 0313 14 31202 44 625 711 84677 921 101

No. 32 80 Mw Aqueous Homogeneous Burner Reactor

ORSORT

Reference: CF-57-8-6.

Originators: R. H. Chapman, H. L. Collins, W. J. Dollard, D. Fieno, J. Hernandez-Fragoso, J. W. Miller, H. von Hollen, and C. V. Wheeler.

Status: Feasibility study; term paper, August 1957.

Details: Thermal neutrons, steady state, burner. Fuel-moderator-coolant: solution of 93% enriched U^{235} as uranyl sulfate in H_2O. Core: a 5 ft diameter sphere of stainless-clad carbon steel with a concentric inlet and outlet to distribute the flow. The fuel solution enters the core at 260°C and, 1750 psia, is heated to 290°C before leaving. Steam is produced in 4 vertical heat exchangers coupled to a common horizontal-type steam drum. Control: through negative temperature coefficient and "automatic demand control." Design power: 20,000 kw(e) with an 80% plant factor.

Code: 0313 13 31201 44 624 711 84677 921 101

No. 33 Aqueous Homogeneous Power Plant (Wolverine)

ORNL

References: CF-57-12-8 Rev., Lane, et al, Fluid Fuel Reactors, pp. 473-9.
Originators: M. I. Lundin, R. Van Winkle, and staff, Foster Wheeler Corporation, Wolverine Electric Cooperative.
Status: Wolverine Electric Cooperative proposed joint participation with AEC, Feb. 1, 1956. The AEC cancelled plans to negotiate contract with Foster--Wheeler Corp. for development and construction of the reactor, Oct. 3, 1957. The AEC then requested ORNL to review the proposed design and to prepare a detailed cost analysis, Oct. 22, 1957. The above was issued by ORNL to fulfill this committment. A conceptual design study was included Dec. 11, 1957. Project was completely cancelled in May 1958 as a result of rise in estimated costs.
Details: Thermal neutrons, steady state, single-region burner. Fuel-moderator-coolant: solution of 90% enriched U^{235} as uranyl sulfate in D_2O. Additional U^{235} added periodically to sustain criticality. Fuel solution circulation, maintained by canned-motor pump, goes from core to U-shaped, shell-and-tube heat exchanger and back to core. Outlet temperature: 300°C; inlet: 260°C. Entire primary system is pressurized to 1900 psia by oxygen. Concentric-inlet and outlet design promotes a very high degree of mixing in the core. Pressure vessel: 6 ft diameter stainless-steel within which are thermal shields cooled by H_2O. Control: no control rods; normal control is presumably through negative temperature coefficient. Facilities for emergency dumping. Design power: 10,000 kw(e); 5,000 kw(e) initially. Second superheater-turbogenerator could be added later to increase output to design power.
Code: 0313 14 31202 44 625 711 84677 921 101

No. 34 Homogeneous Fuel Natural Convection Power Reactor

General Electric Company

Reference: GEAP-2 Del.

Originator: Staff, General Electric Co., Atomic Power Equipment Department.

Status: Conceptual design, May 31, 1955; no further work.

Details: Thermal neutrons, steady state, burner. Fuel-moderator-coolant: solution of UO_2SO_4 in H_2O under 1300 psi pressure with 93.5% enriched U^{235} (20-23 gm/l). Core: 26 in. ID hemisphere 13 in. high with a cylinder, 11.75 in. high, attached to the top. Solution temperature increases from 250°C to 280°C in the core; fuel rises by natural convection in the central riser until it reaches the top header. From there it flows down in titanium tubes of the straight tube-and-shell-type heat exchanger that forms the body of the reactor vessel. Heat is removed by natural convection to boil water on the secondary side of the vessel. Pressure vessel: Ti-lined carbon steel that is also a reflector. Control: reactor is self-regulating. Power level: 400 kw(t); 100 kw(e).

Code: 0313 13 31201 44 624 711 84677 921 101

No. 35 80,000 Kwe Homogeneous Reactor Plant

Westinghouse Electric Corp.

Reference: WIAP-9.

Originator: Industrial Atomic Power Group, Westinghouse (J. M. Stein, nuclear; A. J. Surowiec, chemical processing; staff of Vitro Corp. of America, Engng. Div.).

Status: Proposal, February 1955; no further work.

Details: Thermal neutrons, steady state, burner. Fuel-moderator-coolant: solution of 93.4% enriched U^{235} as uranyl sulfate in D_2O. Fuel flows from the core to gas separators to recombiner dump tanks and back to the primary loop of the reactor. Inlet temperature: 482°F; outlet: 572°F. System pressure: 2000 psia. Pressure vessel: 10 ft OD, of 6 in. thick carbon steel; contains thermal shields lined with titanium to make the core region 7 ft ID. Control: negative temperature coefficient and changing fuel concentration. Power rating: 80,000 kw(e).

Code: 0313 14 31202 44 625 711 84677 921 101
83779

No. 36 Enriched Pile

Clinton Laboratories

Reference: Unpublished report.

Originator: L. W. Nordheim.

Status: Proposal; calculations, Nov. 16, 1944.

Details: Thermal neutrons, steady-state, breeder. Fuel-moderator: 800 g U^{235} salt dissolved in D_2O to make 12.5% solution, or 500 gm plutonium salt dissolved in 100-200 liters D_2O. Fertile material: Th^{232} with U^{233} and U^{238} with plutonium. Reflector: graphite. Solution circulated to external heat exchanger. Temperature difference of 50°C should yield 1 Mw or more. At least 0.59 g U^{233} produced per day.

Code:	0312	14	31202	45	625	7X6	84677	921	101
				46		7X2	8XXXX		

No. 37 Pu^{239} Homogeneous Power Reactor

ORNL

Reference: ORNL-527, p. 10.

Originator: C. B. Graham *et al*.

Status: Considered a long term objective, 1949.

Details: Thermal neutrons, steady state, breeder (could be converter). Few details are given regarding this proposal, but they would undoubtedly be very similar to the HRE-1. Fuel-moderator: solution of Pu^{239} salt in D_2O.

Code:	0312	14	31202	46	625	742	84677	921	101
	0311					711			

No. 38 Plutonium-Producer Power Reactor

ORNL

Reference: Unpublished report.

Originator: ORNL, Long-Range Planning Group.

Status: Conceptual design, 1950.

Details: Thermal neutrons, steady-state, one-region breeder. Fuel-moderator: slightly enriched uranyl sulfate solution in D_2O, 91 grams per liter of solution. Fuel enrichment: 0.8 isotopic percent U^{235}. Core contains 175 metric tons of D_2O and 186 tons of uranium. Solution enters reactor through tangential inlets at equator and leaves through a pipe at the top and bottom of the shell. It then flows in high-pressure (1000 psi) steel pipes through an 8 ft shield surrounding the reactor and into heat exchangers in cells surrounding the shield. Inlet temperature of the solution: 208°C; outlet: 250°C. Reactor vessel: stainless-steel sphere, 24 ft diameter, with 5-1/4 in. thick walls. Control: adjusting concentration to make the solution just critical, negative temperature coefficient of the solution, and emergency dumping. Power: 1000 Mw(t); 230 Mw(e).

Code:								
0312	14	31202	42	625	743	83779	921	101
						84677		
						83179		

No. 39 Large Scale Homogeneous Reactor (LSHR)

ORNL

Reference: CF-52-8-7.

Originators: R. H. Ball, A. M. Hallene, and R. J. March.

Status: Economic and conceptual design study, 1952.

Details: Thermal neutrons, steady state, one-region converter. Fuel: homogeneous solution of 1.075% enriched uranyl sulfate, 250 g/l, dissolved in D_2O. Core contained in 16 ft ID pressure shell, 4 in. thick. Thermal shield: three inner shells, 1-1/4 in., 1 in., and 3/4 in. thick, respectively for thermal shield. All shells of carbon steel except innermost, which is Type 347 stainless steel. Core operates at 250°C and 1000 psia. Reactor core surrounded by 5 cells, each containing 2 heat exchangers (steam generators) and 1 turbogenerator. Working fluid in secondary system: H_2O.

Code: 0311 14 31202 42 625 743 84677 921 101

No. 40 Heavy Water Moderated U^{233} Pile

Clinton Laboratories

Reference: CF-3199.

Originator: Laboratory staff.

Status: Preliminary design, 1945.

Details: Thermal neutrons, steady state, two-region breeder. Fuel: solution of sodium uranyl carbonate, 4 g U^{233} per liter. Fertile material: thorium metal in D_2O slurry. Solution circulates to and from heat exchanger. 3 tons D_2O and 12 kg U^{233} needed. Beryllium tank. Power: 100 Mw. Breeding gain per day: 13 gm U^{233}.

Code: 0312 14 31302 44 635 756 84677 92 101
15

No. 41 U^{235} Pilot Plant for 23 Thermal Breeder

Clinton Laboratories

Reference: CF-3199.

Originator: Laboratory staff.

Status: Preliminary design, 1945.

Details: Thermal neutrons, steady-state, two-region breeder. Fuel: solution of enriched sodium uranyl carbonate in D_2O, 1 g U^{235} per liter. Fertile material: thorium metal. Solution circulated to heat exchanger at 200 liters per second. Spherical core tank, 12,000-15,000 liters. Neutron flux: 10^{14} neutrons/cm^2/sec. Power: 10 Mw.

Code: 0312 14 31202 44 625 726 84677 931 101

No. 42 Heavy-Water Moderated U^{235} Solution Pile

Clinton Laboratories

Reference: CF-3352, pp. 2-3.

Originator: Laboratory staff.

Status: Conceptual design, October 1945.

Details: Thermal neutrons, steady state, two-region converter. Fuel: solution of U^{235} salt (probably sodium uranyl carbonate) in D_2O. Moderator: D_2O in solution and surrounding core as reflector. Fertile material: thorium rods in row around D_2O reflector. Reflector: cylindrical graphite surrounding thorium. Solution at 50°C when it enters reactor; at 67°C when it leaves. Control: control rod, to shut and open pile, in thimble in center of reactor; Cd curtain in D_2O reflector also considered; fine control by rods in reflector. Conversion yield: about 0.65.

Code: 0311	14	31202	44	624	726	84677	941	101
						8111X		
						81142		

No. 43 Thermal Breeder for Uranium-233

Clinton Laboratories

Reference: Unpublished report.

Originator: E. P. Wigner.

Status: Proposal, May 1945.

Details: Thermal neutrons, steady state, two-region breeder. Fuel-moderator: U^{233} as sodium uranyl carbonate dissolved in D_2O, 8 gm U^{233} per liter. Excess Na_2CO_3 hinders decomposition of salt. Fertile material: blanket of ThO_2 slurry, 100 g Th per liter, in D_2O. Blanket, 50 cm thick, in Pb-lined tank, is surrounded by 20 cm thick graphite reflector. Solution and slurry pumped to and from separate heat exchangers and through gas separators and other auxiliary equipment. Core vessel: lead-lined Al cylinder, 90 cm diameter and 90 cm high. U^{233} in reactor: 4 kg; total in system: 13-16 kg. D_2O in reactor: 2 tons; total in system: 13-18 tons. Breeding ratio: 1.23.

Code: 0312	14	31202	45	624	756	84677	941	101
						8XXXX		

No. 44 Breeder with Circulating Uranium

Clinton Laboratories

Reference: Unpublished report.

Originators: E. P. Wigner, A. Weinberg, and G. Young.

Status: Proposal, December 1944.

Details: Thermal neutrons, steady state, two-region breeder. Fuel-moderator: solution of U^{233} dissolved in D_2O as nitrate, sulfate, or fluoride. Using U^{233} solution helps eliminate Xe^{135}. Nitrate is preferred because of less decomposition and corrosion. About 3 kg U^{233} needed in reactor. Fertile material: ThO_2, 1 g/cm^3 D_2O. Reflector: thorium and D_2O in equal parts, 50 cm thick layer. Solution pumped into the 400-liter spherical reactor at the bottom, leaves at the top, passes through heat exchangers and gas-removal systems, and returns to the reactor. Outlet temperature: 50°C higher than inlet. Solution flow: 500 liters per second. Power: estimated at 100-200 Mw(t).

Code: 0312 14 31202 45 625 756 84677 941 101
8XXXX

No. 45 Homogeneous Thermal Carbonate Breeder

Clinton Laboratories

Reference: Unpublished report.

Originator: E. P. Wigner.

Status: Proposal, 1945.

Details: Thermal neutrons, steady state, two-region breeder. Fuel: sodium uranyl carbonate, about 4 g U^{235} per liter. Fertile material: thorium blanket. Solution required: 300-350 liters for 5000-kw reactor. Fuel circulates to and from heat exchanger. Spherical reactor.

Code: 0312 14 31202 44 625 756 84677 931 101
8XXXX

No. 46 Homogeneous Thermal Breeder

Clinton Laboratories

Reference: Nuc. Sci. & Eng., 6, No. 5, pp. 421-2, November 1959.

Originators: E. P. Wigner, W. H. Zinn, A. M. Weinberg, G. Young, L. Szilard, and others not mentioned.

Status: Proposal developed during World War II.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator: solution of U^{233} as a salt, probably sulfate, in D_2O. Core tank: U-shaped tube, 100 cm in diameter, with cooling tubes that extend into the U-tube from both sides, leaving the bottom portion of the U free for chain reaction to take place. Fertile material: Th-D_2O blanket surrounding U-tube at the space free of cooling tubes. Pistons in both legs of the tube move solution alternately from left to right and back so that liquid comes in contact with cooling tubes. Solution does not fill up the tube to the pistons; this space is filled with a gas (presumably helium) to prevent contact of the solution and pistons. Control: design does not include a control system but temperature control can be assumed. Power level: none indicated.

Code: 0312 14 31202 45 625 7X6 84677 941 101

No. 47 U^{233} Power Breeder

ORSORT

Reference: CF-51-8-213.

Originators: J. Bick, J. M. LaRue, P. N. Haubenreich, D. D. Rauch, T. Williams, and J. Putnam.

Status: Conceptual design and feasibility study; term paper Aug. 6, 1951.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator-coolant: solution of U^{233} as UO_2SO_4 in H_2O. Breeding blanket: slurry of ThF_4 in D_2O, 78.23 cm thick. Core temperature: 230°C, blanket, 200°C. Pressure of both: 1000 psi. Core vessel: zirconium tank, 139.5 cm radius, with two inlets and two outlets, each with several annuli. A novel feature of this concept is that no slurry passes through the pump in the main slurry flow line. The slurry is separated from the D_2O in a cyclone separator; the D_2O goes through a heat exchanger and then through the pump, then D_2O is recombined with the slurry and both return to the blanket. Reflector: blanket. Control: mainly by negative temperature coefficient; center pipe of Zr running completely through reactor is thimble in which 1 regulating and 1 safety/shim rod travel; neither is described further. Total U^{233} production: 715.6 gms per day. Power: 500 Mw(t); 467 Mw(t) in core and 33 Mw(t) in blanket.

Code: 0312 13 31201 45 624 756 84679 941 101
8111X

No. 48 Isotope and Power Producer

ORSORT

Reference: Unpublished report, ORSORT, August 1951.

Originators: W. E. Edwards, G. E. Garker, F. D. Orazio, P. O. Nadler, L. H. Thacker, P. N. Wood, W. E. Unbehaun, and D. T. Bray.

Status: Conceptual design; term paper, August 1951.

Details: Thermal neutrons, steady state, two-region converter. Fuel-moderator: uranyl sulfate solution, enriched to 1.35% U^{235} in D_2O. Fertile material: slightly enriched uranium solution; Al-Li alloy shot in 6-in. thick layer. Reactor solution flows through a gas-recombination system and a main heat exchanger, to produce steam that is superheated by heat from the recombination system. Blanket cooling: helium at an inlet temperature of 150°F and outlet temperature of 650°F. Helium is cooled by H_2O in a heat exchanger. The primary heat-exchange system has eight separate loops. Control: vertical concentric rods--shim and regulating--boron-steel sleeve; control of solution concentration; negative temperature coefficient; and emergency dumping. Biological shield: barytes concrete. Power: 1360 Mw(t); power density: 40.5 kw per liter.

Code:								
0311	14	31202	42	625	743	8111X	931	101
					788	81111		
						83779		
						84677		
						83179		

No. 49 Ultimate Homogeneous Reactor

ORSORT

Reference: CF-54-8-239.

Originators: R. A. Thomas, J. F. Brunings, L. S. Hall, C. Michelson, J. F. Parrette, L. R. Pletke, R. L. Whitelaw, and W. A. Wittkopf.

Status: Conceptual design; term paper, August 1954.

Details: Thermal neutrons, steady state, two-region breeder. Fuel-moderator-coolant: uranyl sulfate in D_2O at U^{233} concentration of about 10 g/l. Natural circulation. Fertile material: boiling D_2O-ThO_2 slurry as 75 cm-thick blanket. Core inlet temperature: 505°F. Pressure: 1850 psia. Blanket temperature: 625°F. Core heat generates steam; blanket heat provides superheat and reheat. Core in Zircaloy shell, blanket in stainless-steel part-cylindrical, part-spherical shell. Control: automatic safety controls and secondary regulating controls effect shut-down by decreasing uranium concentration. Total power: 200 Mw(t); 44 Mw(t) supplied by blanket. Core 90 cm diameter.

Alternate blanket: H_2O pellet bed blanket may be substituted for boiling D_2O-ThO_2 slurry. Blanket of ThO_2, H_2O, steam; fluidized pebble bed. Pellet spheres 1/16 in. containing U^{233} and ThO_2. Controls same as for slurry; increased U and Th inventory to obtain same blanket power as for slurry.

Code: 0312 14 31202 45 625 756 83779 941 101
786 84677

No. 50 High Temperature Homogeneous Reactor

ORSORT

Reference: CF-55-8-191.

Originators: A. Hauspurg, J. M. Finan, J. W. Geiser, B. H. Hamling, J. J. Happell, K. Moore, and G. R. Thomas.

Status: Design and feasibility study; term paper, August 1955.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator-coolant: solution of $U^{233}O_3$ and CrO_3 in H_2O at 3500 psia. Breeding blanket: slurry of ThO_2 in D_2O. Spherical zirconium core: about 5 ft ID. Blanket container: steel, 10 ft ID; blanket slurry is 2 ft thick. Solution inlet temperature: 750°F; outlet: 1200°F. System is essentially reflector-moderated. Control: negative temperature coefficient. Breeding ratio: 1.003. Power: plant using two such reactors to produce a total of 690 Mw(t), 290 Mw(e).

Code: 0312 13 31201 45 624 756 84677 931 101

No. 51 Two-Region Homogeneous Without a Core Tank

ORNL

Reference: CF-54-6-180.

Originator: S. Visner.

Status: Proposal, 1954; no further work.

Details: Thermal, steady state, breeder. Fuel: uranyl sulfate solution; moderator-coolant: not given, but probably D_2O or H_2O. Slurry blanket: ThO_2, also probably in water. U^{233} is produced. Both fuel and fertile material are located in the same 6-ft-diameter vessel. By use of a rotational flow pattern, the reactor vessel is operated as a centrifugal-type separator to locate the solution in the center and the slurry at the periphery. However, some fuel may be with the slurry, and the density distribution of the slurry may not be distinct. Two designs are given for the core and flow pattern: 1) A fine slurry is recirculated from the outlet in the wall of the cylindrical vessel. After heat is removed, it is injected tangentically at the opposite end in a direction so as to reinforce the rotational flow pattern. Direction could also be reversed to top to bottom. Solution is circulated separately. 2) A coarse slurry may be employed, for which the entire separation can be achieved near the inlet on each pass in one second. One main circulating system is used in either of two ways: slurry and solution could mix at the outlet, pass through the heat exchanger together, and separate at the inlet; or the solution alone could be circulated and the slurry kept in the reactor. Flow velocity for both designs (1 and 2): 20 ft/sec. No other data given.

Code: 0312 1X 312XX 4X 62X 756 84677 931 101

No. 52 K-49 Reactor

ORNL

Reference: Unpublished report, March 11, 1954.

Originator: HRP staff.

Status: Calculations, March 1954; project terminated, 1961.

Details: Thermal neutrons, steady-state, two-region breeder. Fuel-moderator: uranyl sulfate solution in D_2O; 93.5% U^{235} in feed. Critical concentration: 2.810 g U per kg D_2O; 1.405 g U^{235} per kg D_2O. Cooling: presumably by circulation to external heat exchangers. Fertile material: 3-ft blanket of uranyl sulfate in D_2O; 342 g U per kg D_2O; 99.75% U^{238}. Average core temperature: 275°C. Average blanket temperature: 250°C. Core vessel: zirconium sphere, 6 ft diameter, 1/2 in. thick, 3200 liters volume. Control: negative temperature coefficient of solution. Total (core plus blanket) Pu produced per day: 433.5 g. Total power: 450 Mw(t).

Code: 0312 14 31202 44 625 745 84677 931 101

No. 53 Thermal Power Breeder

ORNL

Reference: Unpublished report, ORNL, 1950.

Originator: ORNL Long-Range Planning Group.

Status: Conceptual design, 1950.

Details: Thermal neutrons, steady-state, two-region breeder. Fuel-moderator: $U^{233}O_2SO_4$ dissolved in D_2O. Fertile material: ThO_2 suspended in D_2O. Core solution and blanket circulated through heat exchangers, the solution to a boiler of a steam power plant and the blanket to the feedwater heater of a power plant. Operating temperature: core, 250°C; blanket, 200°C. Spherical core. Three values of ratio: (cm^3 D_2O/g U^{233}) give different reactor characteristics:

cm^3 D_2O/g U^{233}	Spherical core radius, cm	Core volume, liters	Power, kw/kg U^{233}
500	73.4	1650	13,450
1000	91.0	3160	24,800
1500	105.0	4860	34,000

Code: 0312 14 31202 45 625 756 84677 931 101
8XXXX

No. 54 Project Dynamo Aqueous Homogeneous Reactor Design

Reference: MIT-5003 Del., pp. 263-74.

Originators: Project Dynamo staff (MIT) used engineering details for this concept, which had been worked out by ORNL, and modified power level and estimated costs.

Status: Engineering and economic analysis, 1953; no further work.

Details: Thermal neutrons, steady state, breeder. Fuel-coolant-moderator: a solution of $U^{233}O_2SO_4$ in D_2O. Blanket: dispersion of ThO_2 in D_2O. Inlet temperature of both fluids: 216°C; outlet: 250°C. Both are under 1000 psia pressure. Fuel flows upward through diffuser screens in the core to a gas separator. From there it is pumped to heat exchangers, to pumps, and back to the core. Blanket enters at the bottom, flows upward in a 2-ft-thick annulus between the Zr core liner and the steel pressure vessel, and then goes to heat exchangers, pumps, and back to blanket. Core vessel: roughly spherical, 6 ft in diameter. Heat exchangers: single-pass, horizontal, shell-and-tube type with water and steam in a floating shell. Control: shim control is by varying fuel concentration; negative temperature coefficient provides self-regulation of reactor. Breeding gain: 0.167 of U^{233}. Three such reactors are designated for a power station. Power of each reactor: 450 Mw(t)--320 Mw(t) in core, 130 Mw(t) in blanket.

Code: 0312 14 31202 45 625 83779 931 101
84677

No. 55 Aqueous Homogeneous Central Station Power Reactor Design No. 1

ORNL

Reference: CF-55-11-35.

Originator: R. B. Briggs.

Status: Conceptual design, 1955.

Details: Thermal neutrons, steady state, breeder. Fuel-coolant-moderator: solution of UO_2SO_4 in D_2O. (No enrichment given.) Fertile material: 2.25-ft-thick blanket, surrounding the 5-ft-diameter core, of ThO_2-UO_3-D_2O slurry. Both fluids circulate, either in the "straight-through" (from bottom to top) flow arrangement or by means of a "concentric inlet and outlet." Operating temperature at the inlet: 482°F; at outlet: 572°F. Pressure: 200 lbs/sq in. The core tank constructed of Zircaloy plate, and the pressure vessel of carbon steel clad with stainless steel. Control: negative temperature coefficient. Total power: 100 Mw(e)--from core, 360 Mw(t); from blanket, 80 Mw(t). (See Data Sheet No. 66 for Design No. 2.)

Code: 0312 14 31202 4X 625 756 84677 931 101

No. 56 Aqueous Homogeneous Power Reactor Utilizing Thorium-Uranium-233

Nuclear Power Group

Reference: NPG-112.

Originators: Nuclear Power Group (American Gas & Electric Service Corp., Bechtel Corp., Central Illinois Light Co., Commonwealth Edison Co., Illinois Power Co., Kansas City Power and Light Co., Pacific Gas & Electric Co., and Union Electric Co.).

Status: Conceptual design and feasibility study, Feb. 1, 1955; no further work.

Details: Thermal neutrons, steady state, two-region breeder. Fuel-moderator-coolant: solution of U^{233} as uranyl sulfate in D_2O. The 2-ft-thick blanket is a slurry of ThO_2 in D_2O; the breeding gain is 0.11. Blanket in spherical annulus between the 6-ft-diameter Zr core vessel and the 11 ft, 4-in. diameter pressure vessel of stainless-steel-clad carbon steel. A 175-ft-diameter, pressure-tight sphere contains the reactor plant. Fuel solution enters at the top of the core through the 24-in. inner pipe at 258°C and exits at 300°C through the annulus between the two concentric pipes forming the inlet and outlet connections to the vessel. After leaving the core, the solution enters a centrifugal gas separator, from which it passes to 6 heat exchangers. It is then returned to the core by 3 canned-rotor circulating pumps. Pressure is 1800 psia. Control: by fuel concentration and negative temperature coefficient; no control rods. Power: three steam generators to produce 180 Mw(e) net, 360 Mw(t) from core and 280 Mw(t) from the blanket for total of 640 Mw(t).

Code: 0312 14 31202 45 625 756 83779 941 101
84677

No. 57 Single-Fluid Two-Region (SFTR) Aqueous Homogeneous Power Plant

NPG and Babcock & Wilcox

Reference: NPG-171 (BAW-7).

Originators: Nuclear Power Group (see Data Sheet No. 56 for member companys) and Babcock & Wilcox Co., Atomic Energy Division.

Status: Conceptual design and feasibility study, July 1957; no further work.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator-coolant: solution of U^{233} as uranyl sulfate in D_2O. Fertile material: blanket of 14 cylindrical assemblies, filled with ThO_2 pellets and constructed of Zircaloy-2 with internal flow baffles, all arranged around the periphery of the core. The fuel solution enters from a ring header below the reactor vessel and circulates up through the ThO_2 pellet blanket. Fuel solution leaves blanket assemblies at the top, flows down through the core region to exit headers from which it passes to four groups of six boiler heat exchangers each. The design calls for a standby cooler, because the blanket must be cooled at all times. System pressure: 1500 psia; inlet temperature: 514°F; outlet: 572°F. No core vessel as such; one vertical, cylindrical pressure vessel, 9 ft 11 in. ID and 11 ft 9 in. long, with ellipsoidal heads contains the active region 5 ft 4 in. diameter, the blanket assemblies (18 in. diameter) and reflector consisting of alternate layers of stainless steel and the fuel solution (9.5 in. thick). Blanket assemblies are rotated incrementally by a hydraulically-operated mechanism to equalize breeding of U^{233} in the pellet beds and to minimize absorption of neutrons by protactinium. Control: varying the fuel concentration. Power: calculations for various power levels are included; NPG and B & W concluded that 150,000 kw(e), 520 Mw(t) would best serve their purposes.

Code: 0312 14 31202 45 625 756 83779 941 101

No. 58 HAR Model

AERE

Reference: Unpublished information.

Originator: C. L. Brown.

Status: Proposal, based on calculations, suggested as basis for further study, August 1955.

Details: Thermal neutrons, steady state, two-region breeder. Fuel: enriched uranyl sulfate dissolved in D_2O, 1 gm U^{233} per liter. Fertile material: slurry of ThO_2 in D_2O, 1 kg ThO_2 per liter. Core vessel: either a right cylinder, 5 ft diameter and 5 ft high, or a sphere, 5 ft, 9 in. in diameter. Fuel solution (at a flow of 33,000 gpm) and the fertile slurry are circulated into the reactor, out of it to separate heat-exchange systems, and back to the reactor. System pressure: 1000 psi. Total D_2O required: 34 tons. Total uranium required: 17 kg to start up reactor plus 20 kg to achieve 0.1% uranium concentration in blanket. Total power: 350 Mw.

Code: 0312 14 31202 44 625 756 8XXXX 931 101
84677

No. 59 Homogeneous Aqueous Reactor

AERE

Reference: Unpublished information.

Originators: AERE staff.

Status: Design in process, March 14, 1956; later suspended.

Details: Thermal neutrons, steady-state, two-region breeder. Fuel-moderator-coolant: uranyl sulfate dissolved in D_2O: 1.3 g U^{233} per liter. Fertile material: blanket (70 cm thick) of ThO_2 suspended in D_2O, 1 kg per liter. Core vessel: spherical zirconium pressure vessel, wall thickness of 1/2 in. and an ID of 10 ft. Core solution and blanket slurry are cooled by circulating, in separate circuits, to outside heat exchangers. Inlet temperature: estimated at 240°C; outlet: 290°C; system pressure: 1500 psi. Total power: 600 Mw.

Code: 0312 14 31202 45 625 756 84677 931 101

No. 60 Swedish Homogeneous Solution or Suspension Reactor

Allmanna Svenska Elektriska Aktiebolaget

Reference: French Patent 808,774.

Originator: Staff members.

Status: French patent granted, July 11, 1960.

Details: Thermal neutrons, steady state, two-region breeder. Alternative fuels: solution of uranyl sulfate, suspension of uranium oxide, or solution of plutonium sulfate, all in D_2O. Alternative fertile materials: ThO_2 slurry, thorium nitrate solution, or uranium oxide slurry, all in D_2O. Thorium oxide slurry used in example. Critical concentration in core: 1 g U^{235} per liter. Concentration ThO_2 in blanket: 1000 g ThO_2 per liter. Core vessel: cylinder, 150 cm diameter, 300 cm high, with volume of 7 m^3. Core solution and fertile blanket circulated to separate heat exchangers. Means provided to exchange material between core and blanket and to separate slurry into solids and liquid. Arrangement claimed to have advantages of both one- and two-region breeders. System pressure: 105 kg per cm^2. Total power (core plus blanket): 550 Mw(t).

Code:								
0212	14	31202	46	625	746	8XXXX	931	101
			41	635	756	84677		
					752			

No. 61 U^{233} Breeder-U^{235} Converter Reactor

ANL

Reference: Nuclear Sci. and Eng., 1, pp. 343-54, October 1956.

Originators: L. I. Katzin and B. I. Spinrad; authors give credit to M. Treshow for originating the design concept.

Status: Conceptual stage, 1954; no further work.

Details: Thermal neutrons, steady state, two-region breeder (with little modification of operating conditions and none of structure, reactor can be operated as a U^{235} - U^{233} converter). Fuel-moderator: solution of U^{233} in D_2O (enrichment not specified for converter), contained in 72 zirconium fuel tubes (1 cm diameter) per fuel unit or subassembly (15 cm diameter), of which there are 80. Tubes are hung from the top grid so that the average height of tubes from the tank bottom is 1 ft. Around the fuel unit is the coolant guide (or sleeve) of thin-wall Zr or Al. The coolant-breeding blanket: slurry of Th^{232} (preferably) in D_2O, flowing into the plenum or header system at the bottom of the tank and up past the fuel tubes extracting heat from the fuel system. At a height of 11-1/4 ft sleeves are slotted so that the slurry coolant flows over into the moderator. It is extracted at the tank bottom through coolant exit points. Core vessel: steel tank, 12 ft in diameter and 13 ft high. Sleeves are located in a 10 ft diameter area to allow a 2 ft reflector area, presumably the D_2O. The outlet and inlet sides of the tubes have a common header box, the bottom coated with neutron-absorbing material (Cd, B, or Au) located 12 ft above the tank bottom. Fuel goes from there to a splash chamber, above which is a xenon trap and D_2O recombiner, then to a pump before returning to the tubes. Control: not described. Power: 220 Mw(t) from the core and 358 Mw(t) from the blanket.

Code:	0311	14	31302	45	625	756	84677	92	106
	0312			4X			8XXXX		

No. 62 600-Mw Homogeneous Slurry Pile

Manhattan District, Corps of Engineers, U.S. Army

Reference: CC-1383.

Originator: Columbia University Group.

Status: Calculations, 1944.

Details: Thermal neutrons, steady state, one-region converter. Fuel: slurry of enriched (0.71 mol % U^{233}) UO_3 in D_2O (200 g per liter). 50 tons D_2O in system. Slurry presumably cooled by circulation to external heat exchanger. Spherical reactor, 420 cm diameter, 37,000 liters volume. Power: 600 Mw(t).

Code: 0311 14 31302 42 635 753 84677 921 101

No. 63 Homogeneous Slurry Pile

Clinton Laboratories

Reference: Unpublished report.

Originator: Laboratory staff.

Status: Proposal, 1945.

Details: Thermal neutrons, steady state, one-region breeder. Fuel-moderator: natural uranium oxide suspended in D_2O, 250 g per liter. Suspension pumped (34,600 gpm) under pressure to stainless-steel heat exchangers for cooling by H_2O. Coolant inlet temperature: 122°F; outlet temperature: 248°F. Power: estimated at 750 Mw(t), if 50 tons of D_2O were used.

Code: 0312 14 31302 41 635 752 84677 921 101
8XXXX

No. 64 Heterogeneous Slurry Pile

Clinton Laboratories

Reference: Unpublished report.

Originator: Laboratory staff.

Status: Proposal, 1945.

Details: Thermal neutrons, steady state, breeder. Fuel: slurry of natural uranium as UO_2 in D_2O, 3 g U per cc. Moderator: D_2O. Reactor vessel: vertical cylindrical shell-and-tube exchanger. Slurry is inside tubes immersed in D_2O moderator. Separation of concentrated slurry from D_2O claimed to give higher multiplication constant than for homogeneous slurry. Reflector: graphite around reactor. Slurry heated by passing through tubes, cooled by external heat exchangers, and returned to reactor. Moderator can be cooled by separate heat-exchange system. Power: probably exceeds 50 Mw(t).

Code: 0312 14 31302 41 635 752 84677 923 104
8XXXX

No. 65 Natural Uranium-Pu Producer Homogeneous Reactor

ORNL

Reference: ORNL-527, p. 10.

Originator: HRP staff.

Status: Considered a long term objective, 1949.

Details: Thermal, steady state, converter. Fuel-moderator-coolant: slurry of natural uranium oxide in D_2O--about 200 g/l with total volume of about 30 m^3. Alternative fuel: solution of UO_2F_2 in D_2O, run at high pressure and operated at 100°C. Concept offers the possibility of being the most economical Pu producer. Total power: 600 Mw(t), from 6 metric tons of uranium.

Code: 0311 14 31302 41 635 752 84677 921 101
625 742

No. 66 Aqueous Homogeneous Central Station Power Reactor Design No. 2

ORNL

Reference: CF-55-11-35.

Originator: R. B. Briggs.

Status: Conceptual design, 1955.

Details: Thermal neutrons, steady state, one-region breeder. Fuel-coolant-moderator-fertile material: slurry of ThO_2-UO_3-D_2O. Core vessel: 12 ft diameter, of stainless-steel-clad carbon steel. Fuel flow may be "straight-through," from bottom to top of core, with perforated plates at the inlet to insure smooth expansion of fuel. In this arrangement, there is essentially no separation at the wall and no recirculation of the fluid in large eddies. Alternatively, fuel flow may be accomplished by means of a "concentric inlet and outlet" system. Inlet temperature: 482°F; outlet: 572°F; operating pressure: 200 psi. A natural circulation steam generator extracts heat from the circulating slurry. Control: temperature coefficient of reactivity. Power: 440 Mw(t), 100 Mw(e). (See Data Sheet No. 55 for Design No. 1.)

Code: 0312 14 31302 4XX 635 756 85779 932 101

No. 67 One-Region U^{238} Converter and Power Reactor

ORNL

Reference: CF-56-1-26, p. 9.

Originator: HRP staff.

Status: January 1956; one of five reactors evaluated for possible large-scale development. Project terminated, 1961.

Details: Thermal neutrons, steady state, one-region converter. Fuel-moderator-coolant: slurry of uranium trioxide and plutonium dioxide in D_2O; 300 g uranium per liter; 5.6 g fissile material per liter. Maximum fuel temperature: 300°C. Reactor pressure: 1800 psia. Total system volume, 46,000 liters. Reactor diameter: 12 ft. Conversion ratio: 0.88. Power: 440 Mw(t); 100 Mw(e) from generating plant.

Code: 0311 14 31302 47 635 753 84677 921 101

No. 68 HRE-4

ORNL

References: ORNL-2654, pp. 45-52; ORNL-2743, pp. 67-9; ORNL-2561, pp. 63-7.

Originators: HRP staff.

Status: Design studies, Oct. 31, 1958.

Details: Thermal neutrons, steady state, burner (could be breeder). Fuel-moderator-coolant: slurry of ThO_2 with enriched $U^{235}O_2$ in D_2O. It flows upward through the spherical core and down through the heat exchanger. A pump would be added if necessary. System is pressurized by boiling the slurry continuously in the high-pressure dump tank. Operating temperature: 275°C (average); pressure: 1500 psia. Core is 4.5 ft in diameter with a volume of 1520 liters. Control: by moderator during normal operation. Design power level: 5 Mw(t).

Code: 0312 14 31302 44 635 756 84677 921 101

No. 69 Slurry-Core, Slurry-Blanket, Power Breeder

ORNL

Reference: CF-59-7-129, pp. 19-20.

Originators: HRP staff.

Status: 1959, conceptual design; project cancelled, 1961.

Details: Thermal neutrons, steady state, two-region breeder. Fuel-moderator: slurry of ThO_2 and highly enriched uranium-235 (presumably as oxide) in D_2O; 200 g Th per liter, 14.3 g U per liter. Fertile material: 2-ft thick blanket of thorium in D_2O (1000 g Th per liter). Fuel slurry circulates through core at 90,000 gpm, entering at 256°C. Core vessel: zirconium-alloy cylinder, 7 ft diameter, 21 ft long. It circulates through six parallel steam-generator and pumping circuits, and returns to the reactor vessel. Steam is produced at 400 psia and 445°F. Core system is pressurized by heating slurry in a pressurizing vessel. Blanket slurry circulates through two parallel circuits, only one of which contains a heat exchanger. Core and blanket pressurizers are connected to prevent excessive pressure difference across the core tank. Conversion ratio: 1.07-1.10. Power: 1140 Mw(t), 333 Mw(e).

Code: 0312 14 31302 44 635 756 84677 931 101

No. 70 Homogeneous Power Producer

ORSORT

Reference: CF-51-8-137.

Originators: B. W. O. Dickinson, K. A. Hub, G. P. Letz, W. A. Redfield, J. N. Renaker, R. A. Wall, and R. J. Beeley.

Status: Design and feasibility study; term paper, 1951; no further work.

Details: Thermal neutrons, steady state, converter. Fuel-moderator-coolant is a natural UO_3-D_2O slurry at 250°C under 1000 psi. Pu produced in core remains as a supplementary fuel source. Reactor core: 27 ft diameter stainless-steel sphere of the multi-concentric shell type, cooled by 1 in. wide layer of circulating H_2O at 800 psi between the first pressure shell and the outer one. Slurry enters at bottom of the reactor in an annulus surrounding the fluid outlet pipe, passes between first and second inner shells of the core, and enters core through system of orifices in the inner shell. Slurry then leaves the reactor through a flared exit tube, which extends 9 ft into the reactor. Fuel is pumped to 16 heat exchangers and then back to the reactor. No reflector other than pressure shells. Control: boron steel rod clad with stainless-steel; concentric with the liquid outlet, the vertical rod is hydraulically operated; other control is by varying the D_2O/U ratio (shim) and by the negative temperature coefficient. Power: 2000 Mw(t), 400 Mw(e).

Code: 0311	14	31302	41	635	752	81141	91X	101
						83779		
						84677		

No. 71 Homogeneous Slurry Reactor

ORSORT

Reference: Unpublished report.

Originators: W. G. Atkinson, et al.

Status: Conceptual design; term paper, August 1951.

Details: Thermal, steady state, breeder. Fuel-moderator-coolant: natural uranium trioxide in D_2O slurry (14.3 g UO_3 to 100 g D_2O). Slurry circulated to 16 vertical tube heat exchangers. System pressurized at 1015 psi with steam. Reactor temperature: 250°C. Two 21-ft reactors combined to produce power; 21.5 tons natural uranium required; 160 tons D_2O. Cylindrical pressure vessel, 21 ft in diameter, high-tensile carbon steel lined with stainless steel. Control: negative temperature coefficient, concentration control, injection of xenon, and Hg-filled tubes; the tubes, which pass through the center of the core, can add Hg or other poison. Reactor produces saturated steam, at 250 psi, which is superheated by combustion of radiolytic gases. Power per reactor: 500 Mw(t). Plutonium produced can be either used as fuel or separated.

Code: 0312 14 31302 41 635 752 84677 921 101
83779
81585
81596

No. 72 Pennsylvania Advanced Reactor

Westinghouse Electric Corp., and Pennsylvania Power and Light Co.

References: Proc. 2nd U.N. Conf. on Peaceful Uses of Atomic Energy, 9, pp. 202-210; Proc. Am. Power Conf., 19, pp. 640-50, 1957.

Originators: Staff members.

Status: Reference design completed December 1958; project discontinued.

Details: Thermal neutrons, steady state, one-region breeder. Fuel: uranium oxide suspended in D_2O, with a uranium and proactinium concentration of 9 g per kg D_2O. Fertile material: ThO_2, suspended with the uranium oxide in D_2O, with a thorium concentration of 251 g per kg D_2O. Moderator and coolant: D_2O. Slurry of uranium and thorium oxides is pumped into reactor vessel. It leaves through four loops to heat exchangers that supply steam to a secondary system for power generation, and is pumped back to the reactor. Slurry enters reactor at 465°F and at a mass flow rate of 16.5×10^6 lb per hr, and leaves at 580°F. System pressure: 2000 psi. Reactor vessel: spherical, 13 ft in diameter. Equipment included for recombining radiolytic gases and reprocessing slurry. Control: negative temperature coefficient and varying slurry concentration; for emergency shutdown, solution is dumped. Power: 550 Mw(t).

Code:	0312	14	31302	44	634	756	84677	921	101
							83779		
							83179		

No. 73 KEMA Reactor I (SUSPOP)

RCN-KEMA Reactor Development Group

References: Proc. 2nd U.N. Conf., 12, pp. 525-528; Philips Tech. Review, 21, No. 4/5, pp. 109-152, March 12, 1960.

Originators: J. J. Went, et al.

Status: March 12, 1960; in operation.

Details: Thermal neutrons, single-region subcritical assembly. Fuel-moderator-coolant: suspension of uranium dioxide (20% U^{235}) in H_2O. Fuel circulates through heat exchanger. Neutron flux maintained by external neutron source. Reactor vessel: stainless-steel cylinder 28 cm diameter, with cone-shaped ends. Reflector: blocks of BeO around the reactor vessel and blocks of graphite surrounding BeO. Control: adjustment of concentration of suspension and heat removal, plus control rods; three safety rods of boron carbide, one of which is control rod; rods, suspended by electromagnets, drop into the reflector; emergency dumping of suspension. SUSPOP is stage in developing homogeneous suspension reactor.

Code: 0333 13 3101 43 634 754 83779 931 101
81111
84677
81379

No. 74 KEMA Reactor BABYPOP

RCN-KEMA Reactor Development Group

Reference: J. British Nucl. Energy Conf., 2, pp. 395-407, October 1957.

Originators: Laboratory staff.

Status: Unknown.

Details: Thermal neutrons, steady state, zero power, converter. Details otherwise the same as for SUSPOP.

Code: 0311 13 3101 43 634 754 83779 931 101
81111
84677
81379

No. 75 250 Kw Aqueous Homogeneous One-Region Suspension Reactor

RCN-KEMA Reactor Development Group

References: Proc. 1st U.N. Conf. on Peaceful Uses of Atomic Energy, 3, pp. 116-120; Proc. 2nd U.N. Conf. on Peaceful Uses of Atomic Energy, 9, pp. 427-440; IAEA Directory, p. 191.

Originators: P. J. Kreyger, et al.

Status: 1958, design; construction of model experiments in progress, 1962. Construction under consideration, 1962.

Details: Thermal neutrons, steady state, converter. Fuel-moderator-coolant: suspension of 20% enriched UO_2 (4% by volume) in H_2O, particle size 4-13 μ. System pressurized with hydrogen to reduce oxidation, water decomposition, and stress corrosion. Slurry pumped through reactor vessel, gas-liquid contactor, gas separator, and heat exchanger. By-pass stream continuously removes portion of slurry for removing small slurry particles, as well as products formed by fission, corrosion, and erosion. Closed-cycle operation. Inlet slurry temperature: 230°C. Outlet temperature: 250°C. Outlet steam pressure: 40 atm. Total system pressure: 60 atm. Reflector: BeO, surrounded by graphite. Circulating suspension can be cooled or heated by a flow of water into a slurry cooler through which the circulating slurry passes. Activated charcoal added to slurry to reduce sorption of insoluble fission compounds on fuel particles or reactor walls. Control: three boron carbide shim safety rods in the reflector; negative temperature coefficient. Power: 250 kw.

Code:	0311	13	31301	43	634	754	81111	921	101
			31101				84677		
							83779		

No. 76 Canadian Two-Zone Slurry Reactor

Reference: CRNE-558 (AECL-249).

Originator: Chalk River Project, Atomic Energy of Canada, Limited.

Status: Calculations, 1953.

Details: Thermal neutrons, steady state, two-region breeder. Fuel: slurry of natural uranium oxide in D_2O, enriched with plutonium. Fertile material: slurry of natural uranium oxide. Reactor consists of two concentric spheres, inner sphere containing the fuel slurry. The space between the spheres, which comprises the bulk of the reactor, contains the blanket slurry of natural uranium oxide. Pressure on slurry: 1000 psi.

Code:	0212	14	31302	47	635	752	8XXXX	931	101
							84677		

No. 77 10 Mw Homogeneous Suspension Reactor (HR-1)

Institute of Nuclear Research, Czechoslovak Academy of Sciences

References: Proc. 2nd U.N. Conf. on Peaceful Uses of Atomic Energy, 9, pp. 441-446; IAEA Proc. Symp. Power Reactor Exp., 1962, pp. 187-195.

Originators: Staff members.

Status: Conceptual design, July 1961; project discontinued.

Details: Thermal neutrons, steady state, converter. Fuel-moderator-coolant: suspension of enriched (15%) U_3O_8 in H_2O. Circulates into core, through heat exchanger, and back to core. Provisions for separating radiolytic gases. Operating temperature: 280°C; design pressure: 120 atm (1764 psi). Suspension flow rate: 65 liters per sec. Cylindrical core vessel, lined with stainless-steel, with 90 cm diameter and 135 cm height. Control: changing concentration of suspension; vertical safety rod; emergency dumping. Power: 10 Mw(t), 2.5 Mw(e).

Code:	0311	13	31301	43	624	754	811XX	921	101
							83779		
							84677		
							81779		

References

1. J. A. Lane, H. G. MacPherson, and Frank Maslan, eds., Fluid Fuel Reactors, Addison-Wesley Publ. Co., Reading, Mass., 1958.
2. R. B. Briggs and J. A. Swartout, "Aqueous Homogeneous Power Reactors," in Progress in Nuclear Energy, Series II, Reactors, R. A. Charpie, et al, ed., McGraw-Hill, New York, 1956, pp. 357-75; Proc. 1st U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 3, pp. 175-87, United Nations, New York, 1956.
3. C. L. Brown and J. B. Morris, "A Survey of Possible Homogeneous Aqueous Reactor Systems," unpublished report, Atomic Energy Research Establishment, Harwell, England, June 19, 1956.
4. Ref. 1, p. 12.
5. L. W. Nordheim, "The Case for an Enriched Pile," CF-44-11-236, Clinton Laboratories, November 1944.
6. Ref. 1, pp. 1-11.
7. "New Piles Meeting, Oct. 16-17," minutes by E. Rabinowitch, CF-3352, Metallurgical Laboratory, University of Chicago, Nov. 29, 1945.
8. "Breeder Pile Discussion, Chicago, June 19-20, 1945," minutes by E. Rabinowitch, CF-3199, Metallurgical Laboratory, University of Chicago, 1945.
9. "Civilian Power Reactor Program. Part III. Status Report on Aqueous Homogeneous Reactors As of 1959," TID-8518, Book 3, USAEC, 1960, pp. VIII, IX, XIII.
10. A. M. Weinberg, "Outline of Program for Design of Pilot Model of Homogeneous Power Reactor," CF-49-7-135, Del., ORNL, July 15, 1949.
11. "Nuclear Reactors Built, Being Built, or Planned in the U.S. as of June 30, 1961," TID-8200, USAEC, Oak Ridge, 1961, p. 1.
12. "Status and Prospects of Nuclear Power--an Interim Survey--Status as of July 1958; A Report of the Technical Appraisal Task Force on Nuclear Power," Publication No. 58-13, Edison Electric Institute, New York, 1958, p. 10.
13. A. M. Weinberg, "The Development of Fluid Fuel Reactors," J. Brit. Nuc. Eng. Soc., 1, p. 35, January 1962.
14. C. E. Winters and A. M. Weinberg, "Homogeneous Reactor Experiment Feasibility Report," ORNL-730, ORNL, July 6, 1950, Decl. March 7, 1957.

15. Directory of Nuclear Reactors, Volume II, Research, Test, and Experimental Reactors, International Atomic Energy Agency, Vienna, 1959, p. 153.
16. Ref. 1, pp. 348-59.
17. C. B. Graham, C. H. Secoy, I. Spiewak, E. H. Taylor, A. M. Weinberg, C. E. Winters, and J. L. English, "Homogeneous Reactor Preliminary Process Design Report," ORNL-527 Del. , ORNL, Dec. 28, 1949; Decl. March 18, 1957.
18. C. L. Segaser and F. C. Zapp, "HRE Design Manual," TID-10082, ORNL, Nov. 18, 1952, Decl. April 5, 1957.
19. HRP Staff, "The Homogeneous Reactor Experiment No. 2," Proc. 2nd U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 9, pp. 509-527, United Nations, New York, 1958.
20. Ref. 1, pp. 359-397.
21. Ref. 9, pp. XIII-XIV.
22. "Homogeneous Reactor Project Reports:" ORNL-2096, May 10, 1956; ORNL-2148 Del. , Oct. 10, 1957; ORNL-2222, Feb. 7, 1957; ORNL-2272, April 22, 1957; ORNL-2379, Oct. 10, 1957; ORNL-3167, Sept. 6, 1957.
23. S. E. Beall and J. A. Swartout, "The Homogeneous Reactor Test," Proc. 1st U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 3, pp. 263-282, United Nations, New York, 1956.
24. E. C. Hise and I. Spiewak, "Repair of the HRE-2 Core," Nucleonics, 19, No. 3, pp. 100-3, March 1961.
25. "Major Activities in the Atomic Energy Programs, January-December 1961," USAEC, January 1962.
26. R. H. Chapman, "HRE-2 Replacement Reactor Study No. 2," CF-61-7-54, ORNL, July 3, 1961.
27. I. Spiewak and F. N. Peebles, unpublished report, ORNL, June 30, 1960.
28. R. H. Chapman, "HRT Replacement Reactor Vessel Study," CF-59-8-110, ORNL, Aug. 5, 1959.
29. H. C. Ott, "Meeting of Evaluation Groups June 22-23, 1953," WASH-148 Del., ANL, Aug. 19, 1953.
30. J. A. Lane, unpublished report, ORNL, Dec. 10, 1951.
31. R. B. Briggs, "Aqueous Homogeneous Reactors for Producing Central Station Power," ORNL-1642, Del., ORNL, May 21, 1954. Decl. Feb. 25, 1957.
32. R. B. Briggs, "Homogeneous Reactors for Producing Central Station Power," CF-54-2-8, ORNL, 1954.

33. I. Spiewak and F. N. Peebles, "Tentative Evaluation of Thorium Breeder Reactors with Pebble Blankets," unpublished report, ORNL, Jan. 4, 1960.
34. "Aqueous Homogeneous Reactors, Second Annual and Final Report to AEC," PNG-7, Pacific Northwest Power Group, Richland, Wash., February 1956.
35. C. J. Borkowski to S. A. Swartout, "Design Review Committee Report on the Homogeneous Reactor Test, (HRT)," CF-55-6-53, ORNL, June 7, 1955.
36. M. Tobias, "Breeding Reactors," CF-55-6-157, ORNL, June 7, 1955, p. 19.
37. "Major Activities in the Atomic Energy Programs, July-Dec. 1954," USAEC, 17th Semi-annual Report, U. S. Government Printing Office, Wash., D. C., p. 24.
38. S. McLain, "Nuclear Power Reactors," _Electrical Eng._, _74_, No. 1, February 1955, p. 144.
39. "Major Activities in the Atomic Energy Programs, Jan.-June 1955," USAEC, 18th Semi-annual Report, U. S. Government Printing Office, Wash., D. C., p. 38.
40. W. E. Thompson, comp., "Homogeneous Reactor Project Quarterly Progress Report for Period Ending May 15, 1951," ORNL-1057, ORNL, Oct. 10, 1951, p. 2.
41. R. B. Briggs, "Design Data for the ISHR," 1st ed., CF-52-8-31, ORNL, 1952.
42. F. C. Zapp and P. N. Haubenreich, "Design Data for the ISHR," 2nd ed., CF-52-11-161, ORNL, Nov. 18, 1952.
43. P. R. Crowley and A. S. Kitzes, "Feasibility of a Fluidized Thorium Oxide Blanket," CF-53-9-94, ORNL, Sept. 2, 1953.
44. Ref. 1, p. 8.
45. W. E. Thompson, comp., "Homogeneous Reactor Project Quarterly Progress Report for Period Ending March 15, 1952," ORNL-1280, ORNL, July 14, 1952, p. VIII.
46. Ref. 1, pp. 408-413; Ref. 43, pp. 144-153.
47. J. P. Sanders and P. N. Haubenreich, "Cooling Tower Reactor," CF-53-10-185, ORNL, Oct. 14, 1953.
48. W. E. Thompson, comp., "Homogeneous Reactor Project Quarterly Progress Report for Period Ending October 1, 1952," ORNL-1424, Del., ORNL, Jan. 10, 1953, pp. 53-57.
49. W. E. Thompson, comp., "Homogeneous Reactor Project Quarterly Progress Report for Period Ending January 1, 1953," ORNL-1478, Del., ORNL, March 3, 1953, pp. 23-40.

50. W. E. Thompson, comp., "Homogeneous Reactor Project Quarterly Progress Report for Period Ending July 31, 1953," ORNL-1605, ORNL, Oct. 20, 1953, pp. 46-47.

51. Ref. 50, pp. 44-45.

52. Ref. 49, pp. 40-41.

53. R. B. Korsmeyer, "Two-Region Thorium Breeder Reactors. Conceptual Design Status," HRP Civilian Power Reactor Conference held at Oak Ridge, March 21-22, 1956, TID-7524, Oak Ridge, March 1957, pp. 65-77.

54. H. F. McDuffie, D. C. Kelly, "Homogeneous Reactor Project Quarterly Report for Period Ending January 31, 1955," ORNL-1853, ORNL, Feb. 16, 1955.

55. R. H. Chapman, "HRE-3 Preliminary Design Summary and Reference Report," CF-58-11-112, ORNL, Nov. 10, 1958.

56. "Homogeneous Reactor Project Quarterly Progress Report for Periods Ending April 30 and July 31, 1958," ORNL-2561, ORNL, Feb. 4, 1959.

57. "AEC Puts Together a Long-Range Power Program," Nucleonics, 18, No. 4, April 1960, pp. 71-82.

58. Ref. 40, p. 1.

59. S. E. Beall, E. G. Bohlmann, R. B. Briggs, F. R. Bruce, J. A. Lane, R. N. Lyon, C. H. Secoy, J. A. Swartout and C. E. Winters, "Status and Objectives--Homogeneous Reactor Project. Summaries of Presentations to the Reactor Subcommittee of the General Advisory Committee," CF-56-1-26, Del., ORNL, Jan. 10, 1956.

60. HRP Staff, "Thermal Breeder Reactor Program--Aqueous Homogeneous Reactors," CF-59-7-129, ORNL, 1959, pp. 11-18.

61. "Homogeneous Reactor Program Quarterly Progress Report for Period Ending January 31, 1960," ORNL-2920, ORNL, April 29, 1960, pp. 27-36.

62. L. G. Alexander, W. L. Carter, R. H. Chapman, R. W. Kinyon, J. W. Miller, and R. Van Winkle, "Thorium Breeder Reactor Evaluation. Part I. Fuel Yield and Fuel Cycle Costs in Five Thermal Breeders," CF-61-3-9, ORNL, May 24, 1961, pp. 51-67a.

63. R. L. Stover, D. J. Burkhardt, and C. A. Negin, unpublished data, MIT Engineering Practice School, Oak Ridge, Oct. 18, 1960.

64. P. C. Zmola, "Comments on Boiling Slurry Blankets for Homogeneous Reactors," CF-55-9-125, ORNL, Sept. 27, 1955. Decl. Feb. 19, 1957.

65. L. D. P. King, "Los Alamos Power Reactor Experiment and Its Assorted Hazards," LAMS-1611, Del., LASL, Dec. 2, 1953.

66. Ref. 1, pp. 397-405.

67. H. S. Isbin, "Catalog of Nuclear Reactors," Proc. 2nd U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 8, p. 568, United Nations, New York, 1958.

68. D. Froman, R. P. Hammond, and L. D. P. King, "Los Alamos Power Reactor Experiments," Proc. 1st U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 3, pp. 283-286, United Nations, New York, 1955.

69. R. A. Clark, "Los Alamos Power Reactor Experiment No. II. LAPRE II," LA-2465, LASL, April 1960.

70. L. D. P. King and R. P. Hammond, "Proposed Specifications for a LAPRE II Type Reactor for DEW NET Applications," LASL Internal Report, K-1-824, LASL, June 1955.

71. "Design Study for Homogeneous Phosphoric Reactor (HPAR)," SC-4459 (RR), Sandia Corporation, August 1960.

72. "500 Mw Mobile Reactor," CF-50-10-114 Rev. , ORNL, Oct. 24, 1950.

73. L. B. Borst, "Atomic Powered Locomotive," Business Opportunities in Atomic Energy, March 15-16, 1954 Meeting, Atomic Industrial Forum, New York City, pp. c-2 to c-15.

74. A. M. Khachaturov, "Russia Sees Atomic Locomotive," Nucleonics, 14, No. 10, p. R-8, October 1956.

75. "Nuclear-Powered Locomotive's Economic Feasibility Questioned by Railroad Men," Nucleonics, 12, No. 3, pp. 78-80, March 1954.

76. "General Design Criteria for USAF Nuclear Power Plant Applications; a Final Engineering Report," GAI-1499 (AS-225792 or RADC-TR-59-132), Gilbert Assoc. Inc., prepared for Rome Air Development Center, Air Research and Development Command, U. S. Air Force, Griffis Air Force Base, New York, June 1959, p. 124.

77. P. R. Clark, W. O. Chatfield, A. E. Cox, J. M. Detwyler, J. A. Murphy, C. W. Nestor, R. R. Roof, and D. H. Walker, "Homogeneous Reactor for Ship Propulsion. Reactor Design and Feasibility Program," CF-54-8-236, Del. , ORSORT, August 1954.

78. D. W. Montgomery, W. J. Dodson, F. F. Kaiser, W. K. Luckow, and T. J. Pashos, "10 Megawatt Aqueous Homogeneous Circulating Solution Reactor for Producing Electrical Power in Remote Locations," CF-53-10-22, ORSORT, August 1953.

79. W. E. Kinney, R. Brodsky, D. Hillis, J. T. Wagner, and T. J. Ward, "A Homogeneous Non-Boiling Natural Convection Power Reactor," CF-53-8-225, ORSORT, August 1953.

80. R. H. Chapman, H. L. Collins, W. J. Dollard, D. Fieno, J. Hernandez-Fragoso, J. W. Miller, H. von Hollen, and C. V. Wheeler, "An 80 Megawatt Aqueous Homogeneous Burner Reactor," CF-57-8-6, ORSORT, August 1957.

81. C. B. Ellis, "The 10,000 Kw Aqueous Homogeneous Reactor," Proc. Annual Conf. of Atomic Industrial Forum, Inc., Forum Report No. 12, II, p. 43, September 1956.

82. M. I. Lundin and R. Van Winkle, "Conceptual Design and Evaluation Study of 10,000 Kwe Aqueous Homogeneous Nuclear Power Plant," CF-57-12-8, Rev., ORNL, Dec. 11, 1957.

83. Ref. 1, pp. 473-479.

84. "Homogeneous Circulating Fuel Reactor Power Plant: Conceptual Design Study Report," GEAP-2, Del., General Electric Co., May 31, 1955. Decl. with del. Feb. 25, 1957.

85. D. H. Fax, "Proposed 80,000 Kilowatt Homogeneous Reactor Plant; Process and Plant Description," WIAP-9, Westinghouse Electric Corp., February 1955.

86. Ref. 34, p. 8.

87. B. A. Mong, J. E. Colgan, R. A. D'Elia, J. S. Mooradian, G. K. Rhode, and P. M. Wood, "A Design Study of a Low-Power Aqueous Homogeneous Boiling Reactor Power Plant," BW-AED-502, Babcock and Wilcox Company, June 1955.

88. P. R. Kasten and H. C. Claiborne, "Fuel Costs in Homogeneous U-235 Burners," Nucleonics, 14, No. 11, pp. 88-91, November 1956.

89. L. W. Nordheim, unpublished report, Clinton Laboratories, Nov. 16, 1944.

90. Ref. 17, p. 10.

91. Long-Range Planning Group, unpublished report, ORNL, 1950.

92. R. H. Ball, A. M. Hallene, R. J. March, "An Economic Study of a 1000 Megawatt Homogeneous Reactor," CF-52-8-7, ORSORT, Aug. 12, 1952.

93. "A Third Report on the Feasibility of Power Generation Using Nuclear Energy," CEPS-1121, Del., CEPS Group of Commonwealth Edison, June 15, 1953.

94. H. C. Claiborne and M. Tobias, "Some Economic Aspects of Thorium Breeder Reactors," ORNL-1810, ORNL, Oct. 27, 1955.

95. H. C. Claiborne and T. B. Fowler, "Fuel Cost of Power Reactors Fueled by UO_2SO_4-Li_2SO_4-D_2O Solution," CF-56-1-145, ORNL, Jan. 30, 1956.

96. "Reactors Developed in the USSR and Its Bloc Countries," AD-248402, Air Information Division, Oct. 31, 1960.

97. R. Oravec, "Survey of Soviet Reactors to September 1958," AECL-1011, Atomic Energy of Canada, Ltd., 1959.

98. Ref. 8, pp. 5-7.

99. Ref. 7, pp. 2-3.

100. E. P. Wigner, A. Weinberg, and G. Young, unpublished report, Clinton Laboratories, December 1944.

101. E. P. Wigner, unpublished report, Clinton Laboratories, 1945.

102. E. P. Wigner, "New Ideas for Nuclear Reactors," Nuc. Sci. and Eng., 6, No. 5, pp. 421-422, November 1959.

103. J. Bick , J. M. LaRue, P. N. Haubenreich, D. Rauch, T. Williams, and G. Putnam, "Reactor Design and Feasibility Problem, U-233 Power Breeder," CF-51-8-213, ORSORT, Aug. 6 , 1951.

104. W. E. Edwards, G. E. Garker, F. D. Orazio, P. O. Nadler, L. H. Thacker, P. N. Wood, W. E. Unbehaun, and D. T. Bray, unpublished report, ORSORT, August 1951.

105. R. A. Thomas, J. F. Brunings, L. S. Hall, C. Michelson, J. F. Parrette, L. R. Pletke, R. L. Whitelaw, and W. A. Wittkopf, "Ultimate Homogeneous Reactor," CF-54-8-239, ORSORT, August 1954, p. 5.

106. A. Hauspurg, J. M. Finan, J. W. Geiser, B. H. Hamling, J. J. Happell, K. Moore, and G. R. Thomas, "High-Temperature Aqueous Homogeneous Reactor," CF-55-8-191, ORSORT, August 1955.

107. Sidney Visner, "A Two-Region Homogeneous Reactor Without a Core Tank," CF-54-6-180, ORNL, June 23, 1954.

108. R. B. Briggs, "Calculations for the K-49 Reactor," unpublished report, ORNL, March 11, 1954.

109. Unpublished report, ORNL, 1950.

110. "Power Plants with Thermal Reactors, An Engineering and Economic Analysis," MIT-5003 Del. , Mass. Inst. Tech., Sept. 15, 1953, pp. 263-74.

111. R. B. Briggs, "Aqueous Homogeneous Reactors for Producing Central Station Power," CF-55-11-35, ORNL, Nov. 8, 1955.

112. W. L. Carter to H. E. Goeller and W. E. Unger, "Preliminary Design Study of Proposed Plutonium Loop," CF-55-3-178, ORNL, March 21, 1955.

113. T. G. LeClair, "A Current Design for a Full-Scale Reactor Power Plant," Nuclear Engineering, Part III, Chem. Engr. Prog. Symposium Series, 50, No. 13, Am. Inst. Chem. Engrs., New York, 1954, pp. 122-8.

114. H. G. Carson and L. H. Landrum, eds., "Preliminary Design and Cost Estimate for the Production of Central Station Power from an Aqueous Homogeneous Reactor Utilizing Thorium-Uranium-233," NPG-112, Chicago Nuclear Power Group, Chicago; Babcock and Wilcox, Lynchburg, Feb. 1, 1955.

115. Sidney Visner, "Nuclear Calculations for Homogeneous Reactors Producing U-233," CF-51-10-110, ORNL, Oct. 22, 1951, Decl. Feb. 14, 1957.

116. P. N. Haubenreich, "Calculations for Thorium- and Uranium-Fueled Reactors," CF-53-12-1, ORNL, Feb. 8, 1954, Decl. Feb. 13, 1957.

117. M. Tobias, P. N. Haubenreich, and R. E. Aven, "Conversion in a Two-Region Reactor," CF-53-2-134, ORNL, Feb. 16, 1955. Decl. Dec. 20, 1955.

118. E. D. Arnold, A. T. Gresky, A. R. Irvine, and R. J. Klotzbach, "Preliminary Cost Estimation: Chemical Processing or Fuel Costs Associated With a Power Station of Three K-23 Reactors," CF-54-7-63, ORNL, July 14, 1954, Decl. Nov. 13, 1957.

119. "Single-Fluid Two-Region Aqueous Homogeneous Reactor Power Plant; Conceptual Design and Feasibility Study. Final Report," NPG-171 (BAW-7), Nuclear Power Group, Chicago; Babcock and Wilcox, Lynchburg, July 1957.

120. C. L. Brown, "A Preliminary Assessment of Two-Zone Heavy Water Homogeneous Systems," unpublished report, Atomic Energy Research Establishment, Harwell, England, August 1955.

121. R. Hurst, I. Wells, and D. Newby, "The Homogeneous Aqueous Reactor," J. British Nucl. Energy Conf., 2, pp. 395-407, October 1957.

122. Unpublished report on a homogeneous aqueous power reactor, Atomic Energy Research Establishment, 1956.

123. P. G. Jones and R. G. Snowden, "Thorium Nitrate Solution as a Breeder Blanket in the H.A.R. Pt. 1, Thermal Stability," AERE C/M298 (HARD(B)/P-26), Atomic Energy Research Establishment, 1956.

124. K. Ishikawa, "Aqueous Homogeneous Reactor," Conference on the Peaceful Uses of Atomic Energy, May 13-17, 1957, Japan Atomic Industrial Forum and U. S. Atomic Industrial Forum, 1957, pp. 86-95.

125. V. Vartosheck, "Utilization of Natural Uranium in a Homogeneous Reactor," Soviet Journal of Atomic Energy, 7, No. 6, pp. 981-6, April 1961.
126. Allmanna Svenska Elektriska Aktiebolaget, "Homogeneous Nuclear Reactor," French Patent No. 808,774, July 11, 1960.
127. L. I. Katzin and B. I. Spinrad, "U-233 Breeder--U-235 Converter Reactor," Nuclear Sci. and Eng., 1, pp. 343-354, October 1956.
128. W. A. McRae and T. L. O'Connor, "Study of a New Fuel Concept for Aqueous Homogeneous Reactors," Q-317, Ionics, Inc., Oct. 21, 1959.
129. Ref. 1, p. 129.
130. I. Kirshenbaum, G. M. Murphy, and H. C. Urey, eds., "Utilization of Heavy Water," TID-5226, Columbia University, New York, 1951, Decl. Feb. 27, 1957; published in National Nuclear Energy Series III-4B.
131. C. F. Hiskey, "Chemical Research. The Heavy-Water Homogeneous Pile: A Review of Chemical Researches and Problems," CC-1383, Columbia University, 1944.
132. Unpublished report, Clinton Laboratories, 1945.
133. Unpublished report, Clinton Laboratories, 1945.
134. C. E. Winters and C. H. Secoy, eds., "Homogeneous Reactor Experiment Quarterly Progress Report for Period Ending November 30, 1950," ORNL-925, ORNL, Jan. 30, 1951, Decl. March 2, 1957.
135. W. E. Thompson, comp., "Homogeneous Reactor Experiment Quarterly Progress Report for Period Ending February 28, 1951," ORNL-990, ORNL, May 18, 1951, Decl. March 1, 1957.
136. W. E. Thompson, comp., "Homogeneous Reactor Project Quarterly Progress Report for Period Ending May 15, 1951," ORNL-1057, ORNL, Oct. 10, 1951, Decl. March 12, 1957.
137. M. W. Rosenthal and M. Tobias, "Nuclear Characteristics of Two-Region Slurry Reactors," CF-56-12-82, ORNL, Dec. 20, 1956.
138. "Homogeneous Reactor Project Quarterly Progress Report for Periods Ending April 30 and July 31, 1958," ORNL-2561, ORNL, Feb. 4, 1959, pp. 63-67.
139. "Homogeneous Reactor Project Quarterly Progress Report for Period Ending October 31, 1958," ORNL-2654, ORNL, Feb. 24, 1959, pp. 45-52.
140. "Homogeneous Reactor Project Quarterly Progress Report for Period Ending April 30, 1959," ORNL-2743, ORNL, Aug. 8, 1959.

141. B. W. O. Dickinson, K. A. Hub, G. P. Letz, T. A. Redfield, J. N. Renaker, R. A. Woll, and R. J. Beeley, "Reactor Design and Feasibility Problem. Power Producer," CF-51-8-137, ORSORT, Aug. 6, 1951.

142. W. G. Atkinson, H. L. Leichter, Jr., G. N. Mahoffen, T. J. Morley, A. M. Rubenstein, M. Shaw, C. M. Wetzel, and L. S. Mims, unpublished report, ORSORT, August 1951.

143. J. J. Grebe, "Why is Dow-Detroit Edison Working on a Fast Breeder Reactor for Power," Nucleonics, 12, No. 2, pp. 13-15, February 1954.

144. Nucleonics, October 1957, p. 19; June 1958, p. 25.

145. W. E. Johnson, S. Bartnoff, D. H. Fax, and E. U. Powell, "Design Considerations for the Pennsylvania Advanced Reactor Plant," Proc. 2nd U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 9, pp. 202-210, United Nations, New York, 1958.

146. W. E. Shoupp, "The Homogeneous Reactor Project--PAR," Proc. Conf. on Peaceful Uses of Atomic Energy, Tokyo, May 13-17, 1957, Japan Atomic Industrial Forum and U. S. Atomic Industrial Forum, 1957, pp. 80-86.

147. W. E. Johnson, D. H. Fax, and S. C. Townsend, "The PAR Homogeneous Reactor Project--Plant Design and Operating Problems," Proc. Am. Power Conf., 19, pp. 640-50, March 1957.

148. "Atomic Power Development at Westinghouse," Power Engineering, 62, pp. 44-50, December 1958.

149. "PAR--Big Homogeneous Dropped," Nucleonics, 17, No. 1, p. 28, January 1959.

150. W. E. Johnson, D. H. Fax, H. J. Garber, G. R. Taylor, "Technology of Aqueous Thoria Slurries for Single-Region Homogeneous Reactors," Proc. 2nd U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 7, pp. 34-8, United Nations, New York, 1958.

151. K. J. de Jong, J. A. H. Kersten, J. J. Went, H. H. Woldringh and J. J. van Zolingen, "A Subcritical Circulating Suspension Reactor," Proc. 2nd U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 12, pp. 525-528, United Nations, New York, 1958.

152. J. A. H. Kersten, J. J. Went, and J. J. van Zolingen, "Physical Aspects of an Aqueous Homogeneous Suspension Reactor," Proc. IAEA Symposium on Power Reactor Experiments, I, No. 21-26, pp. 197-223, IAEA, Vienna, 1962.

153. J. J. Went, "Reasons Behind the Choice of a Homogeneous Suspension Reactor," Phillips Tech. Review, 21, No. 4/5, pp. 109-21, March 12, 1960. (Part I of "Instrumentation for a Subcritical Homogeneous Suspension Reactor," pp. 109-156.)

154. Ref. 15, 3, pp. 190-191.

155. P. J. Kreyger, Th. van der Plas, B. L. A. van der Schee, J. J. Went, and J. J. van Zolingen, "Development of a 250 Kw Aqueous Homogeneous Single Region Suspension Reactor," Proc. 2nd U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 9, pp. 427-40, United Nations, New York, 1958.

156. H. de Bruyn, M. E. A. Hermans, Th. v. d. Plas, B. L. A. v. d. Schee, and J. J. Went, "The Design of a Small Scale Prototype of a Homogeneous Power Reactor Fueled with a Uranium Oxide Suspension," Proc. 1st U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 3, pp. 116-120, United Nations, New York, 1955.

157. J. J. Went and H. de Bruyn, "Liquid-Fuel Reactors with Uranium Oxides," Nuclear Engineering, Part II, Chem. Engr. Progress Symposium Series, 50, No. 12, Am. Inst. Chem. Engrs., New York, 1954.

158. Euratom Fifth General Report, European Community Information Service, Brussels, April 16, 1962.

159. C. L. Brown and J. B. Morris, Ref. 3, p. 16.

160. W. G. Clarke, "A Study of Single-Zone U-233 Thoria Slurry Reactors," AERE-R/R-2537, (HARD-(A)/P-48), AERE, Harwell, 1958.

161. M. F. Duret and I. L. Wilson, "A Two-Zone Slurry Reactor," AECL-249, (CRNE-558), Atomic Energy of Canada, Ltd., Dec. 17, 1953, Decl. Nov. 9, 1955.

162. V. Zajíc, J. Pfann, and J. Gregor, "Design of an Experimental 10 Mw Homogeneous Reactor Fueled with Circulating Uranium Oxide Suspension in Light Water," Proc. 2nd U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 9, pp. 441-446, United Nations, New York, 1958.

163. "Hydrodynamical Aspects of a Homogeneous Suspension Reactor," Proc. IAEA Symposium on Power Reactor Experiments, I, IAEA, Vienna, 1962, pp. 187-95.

164. V. Zajíc, "The Development of Homogeneous Reactors and the Experience with Czechoslovak Project," Jaderná Energie, 7, pp. 296-300, 1961.

165. "Terminal Report to U. S. Atomic Energy Commission for December 20, 1954, to July 9, 1955," AECU-3197, Atomic Power Equipment Division, General Electric Company, Dec. 2, 1955.

166. "Report of the Fluid Fuels Reactor Task Force to the Division of Reactor Development, United States Atomic Energy Commission," TID-8507, USAEC, February 1959.

167. Unpublished report, ANL, Dec. 18, 1952.

168. J. A. Lane, "Objectives and Potential of Aqueous Homogeneous Reactors," D_2O-Moderated Power Reactors, A Symposium, March 3-5, 1959, TID-7575, USAEC, August 1959, pp. 84-91.

169. J. E. Jones, "HRT Mockup Runs, CS-25 and CS-26," CF-61-4-96, ORNL, April 26, 1961.

Additional References

J. S. Leonard and I. H. Mandil, "An Analysis of a Steam Power Cycle as Applied to a Large Homogeneous Circulating Fuel Reactor," CF-51-1-20, ORNL, Jan. 11, 1961, Decl. Feb. 14, 1957.

S. E. Beall and C. E. Winters, "The Homogeneous Reactor Experiment," Chem. Eng. Progress, 50, No. 5, pp. 256-62, May 1954.

H. C. Ott, "The Homogeneous Natural Uranium-D_2O Pile as a Source of Power," TID-10092 (MonC-374), Clinton Laboratories, January 1948, Decl. Mar. 7, 1957.

I. Kirshenbaum, G. M. Murphy, and H. C. Urey, ed., Utilization of Heavy Water, USAEC, Oak Ridge, 1951.

R. B. Briggs, "Developments in Reactors with Mobile Fuels," Nuclear Congress, New York City, June 4-7, 1962, Preprint No. 30, Vol. I, Engineers Joint Council, New York, 1962.

C. J. Borkowski to J. A. Swartout, "Design Review Committee Report on the Homogeneous Reactor Test (HRT)," CF-55-6-53, ORNL, June 7, 1955. Decl. Feb. 16, 1957.

Chapter 4 Boiling Reactors for Power and Breeding

In this chapter boiling reactors are defined as those in which the heat of fission boils an aqueous solution or slurry in the core, so that the rising steam condenses on a heat exchanger and returns to the core. Here again, confusing nomenclature makes it necessary to distinguish these reactors from several others of similar names. "Water Boilers," as described in Chapter 2, ordinarily do not boil: the boiling appearance is due to evolution of radiolytic gases. These reactors are cooled by circulating the fuel solution. Such boiling-water reactors as the Experimental Boiling Water Reactor are not homogeneous, and do not have fuel in solution. Also excluded are those homogeneous aqueous reactors in which boiling takes place in the blanket and not in the core. These reactors are discussed in Chapter 3.

The boiling reactors in this chapter are of three general types: one-region solution; one-region slurry; and two-region slurry. Also included because of its similarity is a reactor fueled with a slurry or solution that passes through tubes in the critical region and then to the heat exchanger.

Lane has listed some of the advantages of boiling as a method of heat removal.[1] The reactor responds more quickly to sudden increases in reactivity, so that power excursions are minimized. Pumps for circulating fuel are not needed. For the same reactor pressure, steam is delivered to a turbine at a higher temperature. Corrosion and radioactive contamination of external systems are reduced because the fuel solution or slurry is not circulated to an external heat-exchange system. Early doubts as to the nuclear stability of boiling solutions have been removed, and experiments indicate that boiling reactors are comparable with non-boiling reactors in power densities obtained. Much experimental work has been carried out on the reduction and consequent precipitation of uranyl sulfate solutions in steel reactors. Titanium-lined equipment and oxygen have been used to solve this problem.

One of the early boiling concepts was that of E. P. Wigner, who, in 1944, proposed an unusual boiling-solution reactor.[2] It consisted of a series of trays, filled with plutonium-salt solution, that are placed above each other. With enough trays, the solution becomes critical and boils. Steam is condensed and returned to the trays. The heat can be recovered as useful power. Simplicity, no outside holdup, and elimination of fission poisons are some of the advantages discussed. Entrainment of plutonium salts, bubbling, and solution instability are among the possible problems.

Fewer concepts have been developed for boiling aqueous homogeneous reactors than for non-boiling ones, and only one has actually been operated in the U.S.--SUPO, a water boiler, was operated under boiling conditions at different times to test the feasibility of boiling reactors and to study variables. A Russian reactor, however, using boiling slurry was constructed, and a French experimental prototype, PHOEBUS--a reactor physics experiment--is scheduled for construction.

Feasibility of Boiling Reactors

Because of the doubts that a boiling reactor would be stable, several preliminary studies were made.

SUPO Experiments

In 1951, after the construction of HRE-1, the Homogeneous Reactor Project group considered removing heat from a homogeneous reactor by boiling the fuel solution rather than by circulating it. Experiments on bulk boiling at atmospheric pressure in a cylindrical tank 1 ft in diameter indicated that power densities up to 5 kw might be achieved.

To determine nuclear stability, a group from Los Alamos and Oak Ridge operated SUPO under boiling conditions in October 1951, thereby indicating that it could be operated safely as a boiling solution reactor at its maximum power density of 4 kw per liter.[3] Experiments were continued in 1952 and 1953 to confirm the stability of operation[4] and to study such problems as power removal,[5] chemical stability,[6] bubble formation,[7] and different core arrangements.[8]

A small feasibility program started early in 1951 with the construction of a simulator to study air bubbles to determine the rate of heat removal and solve certain control problems.[9] Later in 1951 after the SUPO experiment, there was postulated a reactor with boiling liquid flowing up the center of the core, liberating part of the vapor at the top surface, and flowing down in an annular region along the wall of the containing vessel. Design of a small boiling homogeneous reactor was begun, with power output to 100 kw (2 kw/liter), and pressures to 150 psi.[10] It was designed for temporary use only, to demonstrate higher power densities and void volume than the SUPO and to give experience in design and operation of a boiling reactor. A vertical thimble design was incorporated[11] and the boiling reactor research program was continued to investigate the controllability and operating levels, as well as the mechanics of actually designing, building, and operating such a reactor.[12]

Stein and Kasten concluded on the basis of developed kinetic equations that boiling reactors appear to be inherently stable, although the power density available may limit their applications.[13] Welton also commented that kinetic equations indicate boiling reactors would be stable.[14]

Kasten reported that stability of boiling reactors should increase with the power level, and it should also increase as average vapor bubble size decreases. The reactor pressure should be as high as possible to improve the power density and yet have sufficient bubble nuclei for good stability.[15]

Teapot

Although the experiments with SUPO had demonstrated the stability of boiling reactors, more work was needed. SUPO had unfavorable geometry for steam release,[16] and, since it had cooling tubes rather than coolant in the moderator-fuel solution, water boiled out of the cooling tubes. This boiling removed reactivity from the reactor and made uncertain the amount of reactivity required to produce various power levels when boiling.[4] Also, high-pressure density measurements would be required.[17]

The preliminary design of a boiling reactor experiment (BRE) was initiated at ORNL, and it was nearing completion by March 1962. This apparatus, the TEAPOT, consisted of an unreflected homogeneous reactor of variable height with a solution of uranyl sulfate (93.4% U^{235}) in water as fuel and moderator. It was to be run under bulk boiling conditions, with nuclear power supplying all the energy for vapor formation, at 150 psi and a thermal power load of 250 kw (4 to 5 kw per liter of solution).[18,19] TEAPOT should not be confused with various heterogeneous boiling reactor experiments leading to BORAX.

In January 1953, the difficulty in preventing reduction and precipitation of the uranium in a boiling uranyl sulfate solution became apparent, and construction of the reactor was postponed until experiments in solution-stability were completed. These experiments, completed in October 1953, indicated that at oxygen concentrations likely to be encountered in a boiling reactor (6 to 7 ppm), uranium would be reduced in solution in contact with stainless steel. Because the metallurgy of titanium or zirconium was not sufficiently advanced to build a reactor with them, it was decided to abandon the boiling reactor experiment and continue experiments on such problems as solution stability and steam separation.[20] Also, work on HRE-1 indicated that construction of a second homogeneous reactor experiment would be preferable.[21]

A study was also initiated for a large-scale homogeneous boiling reactor to exist only on paper (Paper Boiling Reactor, PBR), initially a power and plutonium producer whose study would uncover problems of constructing an actual boiling reactor. A 6-ft reactor to operate at one atmosphere was planned, in addition to TEAPOT.[12] Late in 1952, however, intermediate design effort on the experiment was reduced and the immediate objective changed from constructing an operating experiment to developing a mockup of the TEAPOT for more complete testing of alternate components. Construction of the building was also to be delayed until after additional studies of the components and characteristics of the mockup could be made.[22]

Other Concepts

Although there has been comparatively little actual operation of boiling-core reactors, several concepts have been developed.

One-Region Solution Reactors

In an ORNL concept, boiling uranyl sulfate solution in heavy water is contained in a cylindrical reactor.[13] The heat of the fission reaction vaporizes the liquid; the generated steam rises through the liquid and vapor spaces and is condensed by heat exchangers. Slurry and other types could be tried also.

The 10 Mw Homogeneous Boiling Reactor, was designed by ORSORT for use in remote locations.[23] The fuel is 93.4%-enriched uranyl sulfate dissolved in ordinary water . It is controlled by the level of fuel in the core. Power removal is predominantly by natural bubble rise and growth.

Another ORSORT concept postulated production of power--1200 Mw(t)--and plutonium.[24] The core is a boiling solution of uranyl sulfate or slurry of uranium trioxide with an enrichment of about 1% U^{235}. The steam-liquid mixture from the reactor flows to external vapor separators and the liquid returns to the core by gravity through external downcomers. Control might be by change in void fraction or by holdup of the solvent moderator or solute fuel. It was concluded that: (a) to be economical, boiling homogeneous reactors must be very large and operate at high pressure and high volumetric steam fraction, suitable only for large power plants; (b) they appear to be feasible except for unsteady-state behavior; (c) the cost is equal to that of non-boiling forced-circulation reactor.

Babcock and Wilcox in 1955 made a design study of a power plant using a solution of uranyl sulfate in light water as fuel.[25] The total thermal output is 1300 kw, and power is removed from the core by boiling the fuel solution. The heat is transferred to the secondary steam system by condensing primary water on the external surface of a bayonet-type boiler and boiling secondary water within the tubes. In addition, vertical coolant tubes extend through the solution to the bottom of the pot. The reactor is controlled automatically by power demand.

Another single-region reactor type that might use solution or slurry fuel is described in a German patent.[26] In the pear-shaped reactor vessel, the solution or slurry boils, and the vapor is condensed on a horizontal finned cooling tube in the upper portion of the reactor.

A Canadian patent describes a reactor with a core of boiling slurry or solution.[27] The core acts as a heat generator and also as a vapor separator. Thus there is a small external circuit so that less fuel and moderator is needed. The chief feature is a method for separating steam from liquid rapidly. The solution or slurry is injected tangentially to the core periphery to produce a vortex movement in the liquid. Steam bubbles are driven toward the center and the vapor escapes at high speeds.

One-Region Boiling-Slurry-Core Reactors

An ORSORT group published a preliminary design and feasibility study for a plutonium power producer, in which the fuel is boiling uranium trioxide slurry, either natural or slightly enriched, in heavy water.[28] Heat is removed by the saturated vapor leaving the reactor shell and passing directly to the turbines or through chemically fired superheaters upstream from the turbines. Control is by increasing or decreasing the reactive volume or changing the moderator-to-uranium ratio.

Another ORSORT reactor yields 500 Mw(t) and produces uranium-233. The fuel is uranium dioxide and thorium dioxide suspended in heavy water. Natural circulation, with internal liquid-vapor cyclone separators and an internal downcomer, is used for power removal. Control may be effected by varying the volume of slurry or heavy water in the core.[29]

In the Boiling Slurry Reactor Experiment (BSRE), or SLURREX, of the Argonne National Laboratory, slurry is circulated, and power is removed by boiling. The purpose was to determine feasibility and to gain information on operating problems.

Slurry flows up through a 24 in. pipe that defines the core region and returns through downcomers to the core inlet. The slurry boils in the core, and steam is separated in a sausage-shaped section at the core outlet.[30] This reactor can be a burner-converter using the U-H_2O-Th or U-H_2O-Pu fuel cycles, or a breeder using the U-D_2O-Th cycle. It can be controlled by either adjusting the slurry concentration or varying the recirculation velocity.[31] Between April and December 1960, an initial hazards analysis had shown that SLURREX could be operated safely near EBR-II at the National Reactor Testing Station in Idaho. Also, because the performance of boiling reactors depends largely on the hydrodynamic characteristics of the reactor system, an atmospheric-pressure mockup of the vessel had been built with fabrication and erection essentially complete.[32]

A French 500 to 1000 kw experimental prototype, PHOEBUS (Pilot Homogeneous Reactor with Ebullition and Suspension), is comparable to SLURREX in that both use a suspension containing uranium dispersed in light or heavy water with heat extracted by boiling in the reactor. In PHOEBUS a cyclone device is proposed for accelerating this heat extraction, whereby the bubbles of vapor move toward the vortex where they are collected. Liquid fuel is injected tangentially at the periphery of the core and dispersed. The fuel could be either a solution or a suspension, although at present the use of a suspension in water having a pH of about 7 is being considered. Many hydraulic problems are involved, such as stability of the vortex shape and proportions, the mechanism for forming the bubbles, and the fluctuations in spatial concentration of the final suspensions. A 500-kw electrical model has answered some of these questions. PHOEBUS itself is conceived as a reactor physics experiment. Light water will be used for moderator, and the fuel will be suspensions of fine glass spheres of uranium, or possibly particles of uranium dioxide fused at a high temperature.[33]

Two-Region Slurry-Core Reactors

A two-region reactor with boiling slurry in both the core and breeding blanket has been briefly discussed in an ORSORT report.[24] The fuel-moderator-coolant is a slurry of uranium trioxide in heavy water, with thorium dioxide slurry in the blanket. The reactor produces 66 Mw(e).

A Russian experimental power reactor located at Ulyanovsk, near the Volga, uses uranium oxides suspended in heavy water as fuel-moderator-coolant, and a thorium suspension as fertile material. The power level is 25-35 Mw(t)[34] and the coolant pressure 700 psia.[34,35] In related Russian concepts for boiling homogeneous reactors, a boiling solution is given as an alternative; and a reflector filled with a boiling suspension of thorium in heavy water, with an external circulating system, was suggested.[36,37]

Miscellaneous

A British patent describes a reactor in which heat is removed from the critical region by using the steam bubbles formed by boiling in the critical region, which has many upwardly extending tubes of small diameter, to circulate the liquid fuel through a heat exchanger.[38] Such fuels as slurries or solutions--enriched uranyl sulfate in ordinary water, for example--might be used. The length and diameter of the tubes distinguish this invention from previous circulating systems.

Status

As with the non-boiling homogeneous aqueous reactors, the future of boiling reactors is uncertain. Few concepts have gone beyond the design stage, and the early interest does not appear to have led to industrial development, in the U.S. at least. Some investigations have been made of comparative costs. In 1953, Briggs reported that differences in costs between boiling and circulating reactors were small.[39]

A similar experimental power reactor, located at Dimitrovgrad, near the [illegible] Volga, used organic moderator [illegible] [illegible] as fuel-moderator [illegible] and other [illegible] [illegible]. [illegible] [illegible] [illegible] [illegible] [illegible] [illegible] [illegible] [illegible], and a [illegible] [illegible] [illegible] [illegible] water [illegible] [illegible] systems [illegible].

[illegible]

A [illegible] patent describes a reactor in which heat is removed from the fissile region by using the [illegible] tube [illegible] by boiling in the [illegible] [illegible] [illegible] [illegible] small diameter [illegible] [illegible] [illegible] [illegible]. Such tubes [illegible] [illegible] in ordinary water, for [illegible]. The length and diameter of the tubes distinguish this invention from previous circulating reactors.

[illegible]

As with the [illegible] boiling homogeneous aqueous reactors, the future of boiling reactors is uncertain. Few countries have gone beyond the design stage, and [illegible] interest [illegible] [illegible] [illegible] development in the U.S. [illegible] [illegible] [illegible] [illegible] [illegible] [illegible]. In 1971, [illegible] reported that differences in costs between boiling and circulating reactors were small.

D A T A   S H E E T S

BOILING REACTORS

No. 1 Boiler-Converter (Tray Pile)

Manhattan Project

Reference: U. S. Patent 3,070,529.

Originator: E. P. Wigner.

Status: Proposal, 1944.

Details: Thermal neutrons, steady-state, converter. Fuel: solution of plutonium salt in H_2O. Solution fills shallow trays placed 50 cm above each other. Trays preferably not of neutron-absorbing material. Area of trays: approximately 1 m^2. If 60 trays are used, system becomes critical and solution boils. Steam is conducted to condensers and water flows back to trays. Reflector: thorium. Pressure: 10 atm suggested to reduce volume of steam. Advantages: simplicity, no outside holdup, elimination of poisoning fission products, very slight change in critical size caused by bubbles; usability of power. Problems: tray material, plutonium entrainment, solution foaming, possible solution instability. Power: 400 Mw(t).

Code: 0311 13 32201 46 624 726 84677 931 109
8111X

No. 2 Teapot

ORNL

References: CF-52-5-226; CF-52-6-78.

Originators: ORNL staff.

Status: Conceptual stage, 1952; abandoned, 1953.

Details: Thermal neutrons, steady state, burner. Fuel: solution of enriched uranyl sulfate (93.4% U^{235}) in H_2O; 59 g U^{235} per liter of solution. Pressure: up to 150 psi. Reactor vessel: stainless-steel cylinder, 6 ft long, 17-1/2 in. diameter, 1/4 in. thick walls, with 18-in. dished heads. Steam leaves reactor and is cooled by internal and external condensers; condensate returns by gravity. Control: Cd or B^{10} rod. Vertical thimble in center facilitates: inserting source for start-up; inserting control rod; kinetic experiments in which rod would be quickly removed; and measuring vertical flux distribution. Maximum power: 250 kw(t).

Code:								
0313	13	32201	44	624	711	81111	921	109
						81112		
						84677		

No. 3 Homogeneous Boiling Reactor Package Power Plant (HBR)

ORSORT

Reference: CF-53-10-23.

Originators: R. J. Rickert, W. C. Gribble, B. A. Mong, R. G. Stater, and J. J. Wesbecher.

Status: Preliminary design; term paper, Aug. 21, 1953.

Details: Thermal neutrons, steady state, burner. Fuel: 93.4% enriched U^{235} as uranyl sulfate. Moderator: H_2O. Core temperature: 482°F. Pressure: 580 psi. Reactor: 6-ft hemisphere lined with Ti sheet. No reflector, but some reflection provided by tank. Control: adjusting fuel level in core to meet power requirements. Power: 10 Mw(t).

Code: 0313 13 32201 44 627 711 83679 921 101

No. 4 Boiling Homogeneous Power and Plutonium Producer

ORSORT

Reference: CF-54-8-238 Del.

Originators: H. J. Kamack, P. J. Flickinger, F. C. Haag, P. B. Haga, R. G. McGrath, B. R. Moskowitz, D. E. Murphy, and R. B. Nicholson.

Status: Preliminary design; term paper, August 1954.

Details: Thermal neutrons, "unsteady-state behavior," converter. Fuel-moderator-coolant: UO_2SO_4 solution or UO_3 slurry in boiling D_2O with Pu oxide as a precipitate. Fuel enrichment: about 1% U^{235}. Core vessel: Ti-lined, carbon steel circular cylinder, 20-ft diameter, 20 ft high. Single-flux region is operated at 1000-1500 psi and 250°C. To correct "unsteady-state behavior," authors suggest a reflector, operation at higher pressure, and use of H_2O instead of D_2O. Fuel flow: by natural circulation; no pumps are necessary. In the primary power-removal system, the steam-liquid mixture from the reactor flows to multiple external vapor separators, from which the liquid returns to the core by gravity through external downcomers. Steam goes to heat exchangers and the condensate circulates through the thermal shield as it returns to the core by gravity. Control: by moderator holdup. The reactor is designed as a power producer--1200 Mw(t), 346 Mw(e)--with plutonium as by-product.

Code:	03X1	14	32213	42	625	743	84677	921	101
			32313		635	753			

No. 5 Aqueous Homogeneous Boiling Reactor Power Plant

Babcock and Wilcox Co.

Reference: BW-AED-502.

Originators: B. A. Mong, J. E. Colgan (Vitro Corp. of America), R. A. D'Elia, J. S. Mooradian, G. K. Rhode, and P. M. Wood.

Status: Design study, June 1, 1955.

Details: Thermal neutrons, steady state, burner. Fuel-moderator-coolant: solution of 93% enriched U^{235} as UO_2SO_4 in H_2O. Solution contained in a 27 in. diameter Ti liner in a 30 in. ID cylindrical pressure vessel of stainless steel with an elliptical head at one end. Full-power operating temperature: 250°C at 600 psia. Steam generated from the boiling solution passes through an entrainment separator to be condensed on a bayonet tube-type heat exchanger in the top section of the reactor vessel. Steam from the bayonet tubes passes through a turbine to generate electricity. Control: mainly by power demand. Design is for a small power plant for possible remote locations. Power: 1300 kw(t), 100 kw(e), plus 400 kw space heating.

Code: 0313 13 32201 44 624 71 84677 921 101
8XXXX

No. 6 Large-Scale Boiling Slurry Plutonium Power Producer

ORSORT

Reference: CF-51-8-84 Rev.

Originators: G. H. Cohen, K. W. Downes, F. R. Grisak, J. H. Hill, V. P. Kovacik, D. G. Ott, and L. C. Widdoes.

Status: Preliminary design and feasibility study; term paper, August 1951.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator-coolant: boiling slurry of UO_3 (0.81% enriched) in D_2O. Ratio of moderator to uranium: 50. Operating temperature: 313°C. Operating pressure: 1487 psi. Reactor vessel: stainless-steel spherical pressure vessel, 30 ft diameter. Reactive volume: lower half of sphere. Core volume: 200,159 liters. Saturated D_2O vapor goes from reactor to turbines or through chemical-fired equipment. Condensed D_2O is pumped directly back to reactor. Shielding: thermal shield--hemisphere of boron-containing metal in lower hemisphere; shield in upper hemisphere--over-lapping baffles to let steam pass. Control: control of slurry concentration and volume; emergency dumping of slurry; xenon poisoning. Power: 805 Mw(t).

Code: 0312 14 32302 42 635 753 83779 921 101
84677
81596

No. 7 Boiling Homogeneous Reactor

ORSORT

Reference: CF-54-8-240.

Originators: H. R. Zeitlin, H. W. Bertini, W. F. Bourgeois, J. N. Calvin, J. R. Engel, G. H. Farbman, O. R. Meyer, and D. P. Ross.

Status: Preliminary design; term paper, August 1954.

Details: Thermal neutrons, steady state, one-region breeder. Fuel-moderator-coolant: ThO_2-UO_2 (U^{233}) in boiling D_2O slurry. Operating pressure: 1000 psi (design 1350 psi). Natural-circulation type. Temperature: 545°F. Core: cylindrical, 17.5 ft diameter, 17.5 ft high. Pressure vessel: hemispherical-headed cylinder, 20.5 ft OD, 50 ft high. Steam-dumping condenser. Control: relief and control valves. Power: 500 Mw(t).

Code: 0312 14 32302 45 635 756 84677 921 101

No. 8 Boiling Slurry Reactor Experiment (SLURREX)

ANL

References: ANL-6148; ANL-6248.

Originators: J. F. Marchaterre and M. Petrick.

Status: Terminal report, December 1960. Criticality expected 20 months after Title I design is approved.

Details: Thermal neutrons, steady state, converter. Fuel: $ThO_2 \cdot UO_{2.5}$ aqueous slurry, 93% enriched U^{235}. Coolant: demineralized H_2O. Natural-circulation boiling reactor. Design pressure: 13.6 atm; temperature: 205°C (operating--10.2 atm and 181°C). Reflector: H_2O. Control: cruciform boron - stainless-steel control rod moving vertically; gas injection into downcomer; change in fuel concentration. Power: 5 Mw(t). Fabrication and erection of the half-scale mockup were essentially complete at the termination of the SLURREX project.

Code: 0311 13 32301 44 634 756 81111 921 101
81596
83779

No. 9 Aqueous Homogeneous Reactor Plant with Boiling Slurry Blanket

ORSORT

Reference: CF-54-8-238, p. 146.

Originators: H. J. Kamack, P. J. Flickinger, F. C. Haag, P. B. Haga, R. G. McGrath, B. R. Moskowitz, D. E. Murphy, and R. B. Nicholson.

Status: "Leading Data" given as appendix to "Boiling Homogeneous Power and Plutonium Producer," term paper, August 1954.

Details: Thermal neutrons, steady state, evidently a breeder. Fuel-moderator-coolant: slurry of UO_3 in D_2O. Fertile material: ThO_2 in D_2O as boiling slurry blanket. Pressure: 1850 psia; inlet temperature in core: 505°F, outlet: 580°F. Breeding ratio: 1.12. Power: 200 Mw(t), 66 Mw(e).

Code: 0312 14 3X302 45 625 756 8XXXX 9X 101

No. 10 Steam Stirred Homogeneous Nuclear Reactor

Reference: British Patent No. 840,740.

Originators: U. S. Atomic Energy Commission.

Status: Patent granted, July 13, 1960.

Details: Thermal neutrons, steady state, burner. Fuel: enriched uranyl nitrate, sulfate, or other salt in H_2O or D_2O, or plutonium. Example uses 0.5 M uranyl sulfate in H_2O, 90% enriched in U^{235}. Moderator: H_2O. Critical mass U^{235}: 900 g; operating mass: 2 kg. Reflector: 60-in. graphite cube. Reactor housing: three regions: critical, heat-exchanger, and gas-recombination. All are inside larger cylinder. Critical region is spherical shell of stainless steel, 1/4 in. thick. It is perforated at bottom so that solution may enter. A bundle of small tubes welded to top of reactor shell connects it to heat exchanger. Tubes are of such a length that most steam bubbles form in tube rather than critical region. The fuel level is just below the top of the tubes. The nuclear reaction boils the solution, which forms steam bubbles. Multiple tubes minimize bubble surges. At optimum circulation and at 100°C operating temperature, tubes should be at least as long as the critical region. For a critical-region diameter and a tube length of 25 cm each, the optimum tube diameter is about 1 cm. Bubbles move upward into the tubes, forcing out the fuel solution that fills them. This solution circulates through heat exchangers, and back to the reaction vessel, where it is again heated. Heat exchanger: series of flat spaced stainless-steel coils forming a cylindrical bundle. Water vapor and radiolytic gases pass through gas-recombination systems and cooling channels. Condensed water returns to the reactor. Control: negative temperature coefficient; control of fuel concentration, B or Cd sleeve; vertical control rod.

Code:	0313	13	32201	44	624	711	84677	921	101
		14	32202		625		83679		
							81141		
							81142		
							8111X		

References

1. J. A. Lane, H. G. MacPherson, and Frank Maslan, eds., Fluid Fuel Reactors, Addison-Wesley Publ. Co., Reading, Mass., 1958, pp. 21-23.
2. E. P. Wigner, "Neutronic Reactor," U. S. Patent 3,070,529, December 25, 1962. Filed 1944.
3. A. W. Kramer, Boiling Water Reactors, Addison-Wesley Publ. Co., Reading, Mass., 1958, p. 31.
4. R. N. Lyon, "Preliminary Report on the 1953 Los Alamos Boiling Reactor Experiments," CF-53-11-210, ORNL, Nov. 30, 1953. Decl. Feb. 14, 1957. (See also ORNL-1221, p. 133 ff.)
5. R. V. Bailey, "Progress Report on Power Removal from Homogeneous Boiling Reactors," CF-53-11-65, ORNL, Nov. 4, 1953. Decl. April 8, 1957.
6. W. E. Thompson, comp., "Homogeneous Reactor Project. Quarterly Progress Report for Period Ending October 31, 1953," ORNL-1653, ORNL, Feb. 25, 1954, Decl. Mar. 4, 1957, pp. 13-18.
7. J. A. Lane, "The Bubble Problem in the Homogeneous Reactor," CF-49-8-97, (ORNL-728), ORNL, Aug. 5, 1949. Decl. Jan. 30, 1956.
8. W. E. Thompson, comp., "Homogeneous Reactor Project. Quarterly Progress Report for Period Ending July 31, 1953," ORNL-1605, ORNL, Oct. 20, 1953, Decl. Mar. 2, 1957, pp. 19-27.
9. Ibid., for period ending May 15, 1951, ORNL-1057, ORNL, Oct. 10, 1951, Decl. Mar. 12, 1957.
10. Ibid., for period ending November 15, 1951, ORNL-1221, ORNL, April 1, 1952, Decl. Mar. 22, 1957, pp. 86, 133.
11. Ibid., for period ending July 1, 1952, ORNL-1318, ORNL, Sept. 19, 1952, Decl. Mar. 2, 1957, p. 70.
12. Ibid., for period ending October 1, 1952, ORNL-1424 Del., ORNL, pp. 13, 20.
13. J. M. Stein and P. R. Kasten, "Boiling Reactors; A Preliminary Investigation," ORNL-1062, ORNL, Dec. 12, 1951. Decl. Mar. 2, 1957.
14. T. A. Welton, "Comments on the Proposed Boiling Reactor Program," CF-51-11-163, ORNL, Nov. 27, 1951.
15. P. R. Kasten to J. A. Swartout, "Boiling Reactors--Variation of Reactor Stability with Operating Power Level and Average Bubble Size," CF-51-11-130, ORNL, Nov. 23, 1951. Decl. Feb. 7, 1956.

16. Reactor Handbook, Vol. II, Engineering, J. F. Hogerton and R. C. Grass, eds., AECD-3646, USAEC, Washington, 1955. Decl. with del. May 1955, p. 698.
17. Ref. 1, p. 21.
18. R. N. Lyon, "Slurry and Boiling Reactor Research," CF-52-5-226, ORNL, May 12, 1952. Decl. April 2, 1957.
19. P. R. Kasten, "Reactor Physics of TEAPOT," Pt. I, CF-52-6-78, ORNL, June 5, 1952; Pt. II, CF-52-6-118, ORNL, June 20, 1952; Pt. III, CF-52-7-38, ORNL, July 16, 1952; Pt. IV, CF-52-8-128, ORNL, Aug. 15, 1952.
20. Ref. 9, pp. 31-32.
21. Ref. 1, pp. 8, 21-22.
22. W. E. Thompson, comp., "Homogeneous Reactor Project. Quarterly Progress Report for Period Ending January 1, 1953," ORNL-1478, ORNL, March 3, 1953. Decl. with del. Feb. 28, 1957.
23. R. J. Rickert, W. C. Gribble, B. A. Mong, R. G. Stater, and J. J. Wesbecher, "A Preliminary Design Study of a 10 Mw Homogeneous Boiling Reactor Power Package for Use in Remote Locations," CF-53-10-23, ORSORT, Aug. 21, 1953. Decl. Feb. 22, 1957.
24. H. J. Kamack, P. J. Flickinger, F. C. Haag, P. B. Haga, R. G. McGrath, B. R. Moskowitz, D. E. Murphy, and R. B. Nicholson, "Boiling Homogeneous Reactor for Producing Power and Plutonium," CF-54-8-238, ORSORT, August 1954. Decl. with del. May 9, 1957.
25. B. A. Mong, J. E. Colgan, R. A. D'Elia, J. S. Mooradian, G. K. Rhode, and P. M. Wood, "A Design Study of a Low Power Aqueous Homogeneous Boiling Reactor Power Plant," BW-AED-502, Babcock and Wilcox Co., June 1955. Decl. April 3, 1959.
26. Kesselwerke AG., "Homogener Siedereaktor, bei dem die Wärmeaustauschflächen innerhalb des Reaktorgefässes angeordnet sind," German Patent DAS 1,027,339, March 4, 1958, (Atompraxis 4, p. 386, October 1958).
27. M. Grenon, L. Berthod, G. Cohen de Lara, M. Delachanal and G. Halbronn, "Nuclear Reactors," Canadian Patent 618,575, April 18, 1961.
28. G. H. Cohen, L. C. Widdoes, K. W. Downes, F. R. Grisak, J. H. Hill, V. P. Kovacik, and D. G. Ott, "A Preliminary Design and Feasibility Study of a Large Boiling Slurry Plutonium Power Converter," CF-51-8-84 Rev., ORNL, Aug. 1, 1951. Decl. March 9, 1959.

29. H. R. Zeitlin, H. W. Bertini, W. F. Bourgeois, J. N. Calvin, J. R. Engel, G. H. Farbman, O. R. Meyer, and D. P. Ross, "Boiling Homogeneous Reactor for Power and U-233 Production," CF-54-8-240, ORSORT, August 1954. Decl. Mar. 14, 1957.

30. G. A. Freund and J. D. Lokay, "SLURREX--Boiling Slurry Reactor Experiment," Nucleonics, 20, No. 2, pp. 74-75, February 1962.

31. M. Petrick and J. F. Marchaterre, "A Preliminary Design Study of a Boiling Slurry Reactor Experiment," ANL-6148, ANL, April 1960.

32. G. A. Freund, J. D. Lokay, G. C. Milak, and J. C. MacAlpine, "Terminal Report on the Boiling Slurry Reactor Experiment (SLURREX)," ANL-6248, ANL, December 1960.

33. J. Beneveniste, J. Bernot, L. Berthod, G. Cohen de Lara, M. Delachanal, D. Eidelman, P. Fontanet, M. Grenon, G. Halbronn, L. Pontes, G. Raspaud, L. Rolland, J. Tachon, and G. Vendryes, "Ideas Concerning a Homogeneous Reactor Project," Proc. 2nd U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 9, pp. 415-26, United Nations, New York, 1958.

34. "Reactors Developed in the USSR and Its Bloc Countries," AD-248402, Air Information Division, Oct. 31, 1960.

35. "World Reactor Chart, Third Edition," Nuclear Power, January 1962.

36. A. I. Alichanow, W. K. Zavoisky, R. L. Serduk, B. W. Ershler, L. J. Suvorow, "A Boiling Homogeneous Nuclear Reactor for Power," Proc. 1st U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 3, pp. 169-74, United Nations, New York, 1956.

37. V. M. Byakov and B. L. Ioffe, "A Homogeneous Natural-Uranium Reactor," Proc. 2nd U.N. Int. Conf. on Peaceful Uses of Atomic Energy, 13, pp. 462-9, United Nations, New York, 1958.

38. "A Steam Stirred Homogeneous Nuclear Reactor," British Patent 840,740, July 13, 1960.

39. R. B. Briggs, "Cost Comparison of Boiling Reactors with Circulating Solution Reactors," CF-53-3-260, ORNL, March 30, 1953.

29. R. E. [illegible], [illegible] Beeston, W. R. [illegible], [illegible], [illegible], Charlie Leigh, Peterson, [illegible], and [illegible], "[illegible] Homogeneous Reactor for Power and [illegible] Production," [illegible], August 1956. [illegible]

30. C. [illegible] and [illegible], "[illegible] Boiling Slurry Reactor [illegible]," Nucleonics, [illegible], pp. 74-75, February 1962.

31. [illegible] and [illegible], "[illegible] Preliminary Design Study of a Boiling Slurry [illegible] Reactor Plant," [illegible], April 1958.

32. [illegible], [illegible], [illegible], and [illegible], "Final Report on the Boiling Slurry Reactor Experiment (SLURREX)," [illegible], [illegible] 1962.

33. [illegible], [illegible], [illegible] Goedkoop, [illegible], [illegible], [illegible], [illegible], "[illegible] Homogeneous Reactor Project," Proc. [illegible] U. N. Intl. Conf. on Peaceful Uses of Atomic Energy, [illegible], United Nations, New York, 1958.

34. "[illegible] Developed in the [illegible] and [illegible]," [illegible], AEC Information Division, Oct. 31, 1960.

35. "World Reactor Chart, [illegible] Edition," Nuclear Power, January 1962.

36. [illegible], [illegible], [illegible], [illegible], "[illegible] Homogeneous Nuclear Reactor for Power," Proc. [illegible] Intl. Conf. on Peaceful Uses of Atomic Energy, [illegible], pp. [illegible], United Nations, New York, 1958.

37. [illegible] and [illegible], "A Homogeneous [illegible]-Uranium Reactor," Proc. [illegible] Intl. Conf. on Peaceful Uses of Atomic Energy, [illegible], pp. [illegible], United Nations, New York, [illegible].

38. "A [illegible] Homogeneous [illegible] Reactor," [illegible], [illegible], [illegible].

39. [illegible], "[illegible] Comparison of Boiling Reactors with [illegible] [illegible] Reactors," [illegible], [illegible].