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Argonne National Laboratory

CATALOG OF NUCLEAR REACTOR CONCEPTS

Part I. Homogeneous and Quasi-homogeneous Reactors

Section VI. Solid Homogeneous (Semihomogeneous) Reactors

by

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

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AEC Research and
Development Report

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Technical Publications Department

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PREFACE

This report is an additional section in the Catalog of Nuclear Reactor Concepts that was begun with ANL-6892 and continued in ANL-6909, ANL-7092, ANL-7138, and ANL-7180. As in the previous reports, the material is divided into chapters, each with text and references, plus data sheets that cover the individual concepts. The plan of the catalog, with the report numbers for the sections already issued, is given on the next page, which is followed by pages listing the concepts included in this section.

Dr. Charles E. Teeter, formerly employed by the Chicago Operations Office at Argonne, Illinois, is now affiliated with the Southeastern Massachusetts Technological Institute, New Bedford, Mass. Through a consultantship arrangement with Argonne National Laboratory, he is continuing to help guide the organization and compilation of this catalog.

We wish to acknowledge the assistance of Miss Ellen Thro in the preparation of this section.

J.H.M.
June, 1966

PLAN OF CATALOG OF REACTOR CONCEPTS

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REACTOR CONCEPTS DESCRIBED IN THIS REPORT (Cont.)

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PART I. HOMOGENEOUS AND QUASI-HOMOGENEOUS REACTORS

SECTION VI. SOLID HOMOGENEOUS (SEMIHOMOGENEOUS) REACTORS

Chapter 1. Introduction

The concepts described in this section are for reactors in which the solid fuel and moderator are in a uniform combination or mixture, e.g., a uniform dispersion of fuel in a moderator matrix. Excluded are reactors in which the fuel and moderator are discrete, as in some closely related ones in which the fuel and moderator form alternate layers or zones. Both these and the solid homogeneous reactors have also been named semihomogeneous or reactors of a mixed fuel-moderator type. The reactors in which the fuel and moderator are discrete will be described in Part II under their respective methods of cooling. Some solid homogeneous reactors have been described in this Catalog in Part I, Section I, Particulate-Fueled Reactors.

Reactors covered in this section have been designed for research and testing as well as for power production and breeding. The research and test reactors are described in Chapter 2, the power and breeder reactors in Chapter 3.

Uranium oxides are a common form of fuel incorporated with the moderator into a fuel element; carbides, alloys, and the metal itself have also been used. In some fuel-moderators, the moderator is impregnated, by soaking, with a solution of a salt such as uranyl nitrate. Subsequent heating converts this to an oxide or carbide. In some elements, particles of the fuel and moderator are cold pressed and sintered or hot pressed. In some concepts no details of the fabrication have been considered. Graphite, beryllium oxide, beryllium, zirconium hydride, and polyethylene have been moderators, with graphite and beryllium being most common for power reactors and polyethylene used in some low-power reactors. Most of the solid homogeneous reactors are gas-cooled. Breeding is chiefly by the Th^{232}---U^{233} cycle.

The advantages of the solid homogeneous reactor were pointed out in 1960 by Charpie and Perry,[1] who stated, "The high temperature homogeneous or semihomogeneous reactors are at once the most difficult and, in all probability, the most rewarding members of the gas-cooled reactor family." The advantages given include: high fuel-moderator volumes and large heat-transfer surface, which lead to high power densities and specific power; high gas pressure made possible by the compact cores; high temperatures of the exiting coolant gas, which might be high enough to permit direct-cycle operation; good neutron economy; and the possibility of low fuel-cycle costs.

Problems cited by these authors as chief difficulties at the time are development of fuel elements with good mechanical integrity and the capacity for retaining fission products, and the need for rapid refueling to take advantage of the high specific power.

Solid homogeneous reactors were among the first gas-cooled types to be considered in the United States. By 1944-45 Farrington Daniels had evolved designs for pebble-bed reactors (see Chapter 1, Section I, Part I) in which the fuel-moderator is pebbles of uranium carbide and graphite. In 1944 he proposed the use of beryllium oxide in a high-temperature pile. He and his co-workers at the Metallurgical Laboratory, University of Chicago, studied this concept. Later, a group at the Clinton Laboratories (now the Oak Ridge National Laboratory) continued development through 1947, when the concept was abandoned.[2] The fuel-moderator in the 1947 concept was a dispersion of uranium oxide in graphite or beryllium oxide, in the form of tubes. For almost ten years after 1947, there was comparatively little work done on this type of reactor in the U.S., partly because of interest in water-cooled and liquid-metal-cooled reactors and also because of the view that the heat-transfer properties of gases would limit power density and specific power. Such developments as successful foreign experience with gas-cooled reactors and the application of new materials led to a renewed interest in the U.S. in gas-cooled reactors, including those of the solid homogeneous type.[3]

Unlike most concepts previously discussed in Part I, the solid homogeneous reactors have been developed into several commercial-size units. These reactors have been considered particularly attractive because of the high steam temperatures that are feasible with the homogeneous fuel-moderator. Prominent current developments in the United States include the Peach Bottom (Pennsylvania) High-temperature Gas-cooled Reactor (HTGR), the Ultra High Temperature Reactor Experiment (UHTREX), and the Experimental Beryllium Oxide Reactor (EBOR). Foreign development has been extensive, in line with the earlier interest in gas-cooled reactors. Programs in the United Kingdom, Australia, and Japan have included solid homogeneous reactors. Several have been built, and others are planned.

References

1. R.A. Charpie and A.M. Perry, Gas-cooled Reactors--A Summary, Gas-cooled Reactors. A Symposium Sponsored Jointly by the Franklin Institute and the American Nuclear Society, Delaware Valley Section, Monograph No. 7, Journal of The Franklin Institute, May 1960, pp. 9-11.

2. C.R. McCullough et al., Summary Report on Design and Development of High-temperature Gas-cooled Power Pile, MonN-383, Clinton Laboratories, Sept. 15, 1947. Decl. May 10, 1957.

3. Ref. 1, p. 4.

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Chapter 2. Research and Test Reactors

Reactors described in this chapter are low-power ones used to give experimental data, primarily through the radiation they produce, and for testing reactor materials. In most, the fuel, an oxide of uranium, is dispersed in the moderator (polyethylene, zirconium hydride, or graphite). Some of these reactors have been widely used.

An early (1952) model of a solid homogeneous research reactor was the Table Model Reactor, described by Biehl, Hetrick, and Bennett.[1] The fuel-moderator consists of very fine particles of uranium dioxide mixed homogeneously with polyethylene and pressed into a mold. The core radius is 11.8 cm. The authors listed some advantages of a solid system over a fluid one: e.g., gas evolution would be avoided if the moderator were sufficiently stable, because the solid material would tend to trap fission fragments while it would not give off appreciable amounts of decomposition gases. Also a smaller critical size, with smaller core size and shielding, would be possible. Tests showed that the uranium-impregnated polyethylene would withstand long operation at high power. Polyethylene has a somewhat higher hydrogen-atom density than does water (7.8×10^{22} nuclei/cm^3 versus 6.7×10^{22} for water).

The Homogeneous Graphite Reactor, described in 1954 by Stelle,[2] was intended both as a research tool and a source of service irradiations to produce isotopes. Its safety features are a negative temperature coefficient and control and safety rods. The fuel, enriched uranium in U_3O_8, is dispersed in graphite. Heavy water circulates around the fuel-moderator as a coolant. A power of 135 kW(t) and an average neutron flux of 10^{12} neutrons/cm^2/sec were planned.

The ANCO-201 reactor (1955) is a compact, low-power reactor designed for training and research.[3] The fuel-moderator is 20% enriched uranium dioxide particles dispersed in discs of polyethylene. The originators, staff members of the Applied Nucleonics Corporation, stated that this reactor could be used with only 450 g uranium-235 and it is safe enough to allow it to be installed in highly populated areas. The maximum thermal neutron flux is 6×10^6 neutrons/cm^2/sec.

The AGN-201 and -211 series of reactors of Aerojet-General Nucleonics closely resemble each other.[4,5] Many have been installed at universities.

In 1956, Biehl et al.[4] described AGN-201 as a compact, low-cost reactor designed for maximum safety and high sensitivity. The fuel, uranium dioxide dispersed in polyethlene, is in the form of discs of different thicknesses.

In addition to the negative temperature coefficient and rods for safety and control, the reactor has an added safety feature. The core is divided into two halves separated by a fuse of polyethylene that has a high uranium-235 content. If a high flux level occurs, the fuse melts, causing the lower half of the core to fall and separate from the other half. Of neglible power, these reactors are cooled by natural convection, but they can be modified to operate at a maximum power of 20 MW(t). The core and a graphite reflector are in a tank of water for shielding.

The AGN-211 reactor[6,7] is similar in many ways to the AGN-201. The fuel is a dispersion of uranium dioxide in polyethylene, but the fuel elements are foot-long sections, with a graphite reflector at each end. The elements are clad with polyethylene to prevent contact of the coolant (water) with the fuel. The reactor has a power of 100 W(t). This reactor was considered desirable for training in nuclear engineering; for example, different arrangements of core lattices might be tried.

The TRIGA reactors (Training, Research, and Isotope Production Reactor, General Atomic) introduced in 1958, have been widely used in the United States and abroad for training, irradiation, and isotope production. There are four models; Mark I, Mark II, Mark III, and Mark F.[8-14] All have fuel-moderator elements of uranium mixed with zirconium hydride, and a coolant-shield of a pool of water. The fuel elements are cylindrical and are clad. Mark I, II, and III have a graphite reflector and can operate either in the steady state or as pulsed reactors. In steady-state operation, these reactors are low power, but in pulsing they can produce 200 MW(t). In the Mark F, specifically designed for pulsing, the graphite reflector is eliminated; it has more fuel elements, which provide higher heat capacity; and it has a large water tank, for more versatility of radiation.[10] A large prompt negative temperature coefficient of reactivity is chiefly depended upon for safety, although control rods are provided. Advantages claimed for the reactors include: economy, versatility, and safety, with no special containment building needed.

An extrapolation of TRIGA technology for a reactor for powering a bathyscaphe, research craft, or power buoy was suggested in 1960.[15]

Two reactors with the same core as the TRIGA Mark I are General Atomic's REGA (Research Reactor, General Atomic) 10-30 and IRGA (Isotope Production Reactor, General Atomic) reactors.[8,13]

In 1959, Thompson and Fahrner[16] described a design for a low-power research reactor in which the fuel-moderator is graphite impregnated with enriched uranium as U_3O_8 and the coolant is heavy water. The fuel elements are clad with aluminum and filled with helium, with concentric tubes in each element for flow of coolant. The power is 200 kW(t). The authors indicated problems would be the large amount of heavy water needed (about 400 gallons); the limitation on temperature by the boiling of the heavy water; and difficulties in design of the cooling system, even at moderate pressures.

Siemens-Schuckertwerke A.G. announced in 1961 plans to build a solid homogeneous reactor as a prototype for a model to be offered for sale.[17] Few details were given. The reactor is fueled and moderated with a suspension of powdered U_3O_8 in polyethylene and has a graphite reflector and shielding of lead and water.

The homogeneous subcritical subassembly[18] designed by staff members at Aerojet-General Nucleonics is included here because it is intended for use in its own right to measure neutron fluxes, rather than as a step in designing another reactor. It has a cylindrical core of a homogeneous mixture of 20% enriched uranium and polyethylene and includes a neutron source. It can be used without a reflector or with one of either graphite or polyethylene.

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DATA SHEETS

RESEARCH AND TEST REACTORS

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No. 1 Table Model Reactor

North American Aviation, Inc.

Reference: TID-2503 (NAA-SR-Memo-352).

Originators: A.T. Biehl, D.C. Hetrick, and G.A. Bennett.

Status: Concept, June 1952.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: 93% enriched U as UO_2 mixed homogeneously with polyethylene. Coolant: ambient air. Reflector: BeO and graphite. UO_2 particles, not more than 10 μ in size, mixed with powdered polyethylene and pressed in a mold. Impregnation density: 50 mg U/cm^3. Core radius: 11.8 cm. Control: negative temperature coefficient of reactivity.

Code: 0313 16 31714 44 5932 711 84677 921 105

No. 2 Homogeneous Graphite Reactor

North American Aviation, Inc.

Reference: Science, 115, pp. 15-21, Jan. 1, 1954.

Originator: A.M. Stelle.

Status: Design, 1954.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: enriched U as U_3O_8, uniformly dispersed in graphite. Coolant: D_2O. Core: blocks of graphite - U_3O_8. D_2O circulates around fuel-moderator. Control: negative temperature coefficient; vertical safety and control rods of stainless-steel tubes filled with B_4C. Reflector: 2 ft graphite around core. Core tank: cylinder 46 in. diam., 93 in. high. Average thermal flux: 10^{12} neutrons/cm^2/sec. Core will last 20,000 hours. Power: 135 kW(t).

Code: 0313 12 31102 4X 5932 711 84677 9XX 105
81X11

No. 3 ANCO 201 Nuclear Reactor

Applied Nucleonics Corp.

Reference: NP-7578.

Originator: Staff members.

Status: Preliminary design, 1955.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: 450 g 20% enriched uranium as UO_2 particles, 10-20 μ, embedded in discs of radiation-stabilized polyethylene. Reflector: graphite. Coolant: ambient air. Core: right circular cylinder, 9 in. diam. and 8¼ in. high, with Al structure. Reflector surrounds core. Control: 2 safety and 1 control rod contain fuel--reactor is subcritical when they are withdrawn; second control rod contains Cd and operates in usual manner. Neutron flux: 6×10^6 neutrons/cm^2/sec. Power: 100 milliwatts.

Code:	0313	16	31714	43	5932	711	84677	921	105
							81112		
							83119		

No. 4 AGN-201- Reactor

Aerojet-General Nucleonics

References: Directory of Nuclear Reactors, III, 1960; Nucleonics, 14, No. 9, pp. 100-103, Sept. 1956.

Originators: Staff members.

Status: AGN-201-

No.	Owner	Location	Startup (dismantled)
-100	Naval Postgraduate School	Monterey, Calif.	1956
-101	Catholic University	Wash., D.C.	1957
-102	Oklahoma A&M	Stillwater	"
P-103	Aerojet-General Nucleonics	San Ramon, Calif.	"
-104	U. of Akron	Akron, O.	"
M-105	National Naval Med. Center New York University	Bethesda, Md. New York	1957 (1962) 1964
-106	Texas A&M	College Station	1957
-107	U. of Utah	Salt Lake City	"
-108	ANL	Argonne, Ill.	"
-109	Colorado State U.	Ft. Collins	"
-110	U. of Palermo	Palermo, Italy	1960
-111	U. of Geneva*	Geneva, Switzerland	"
-112	U. of Calif.	Berkeley	1957
-113	U. of Delaware	Newark	1958
-114	Oregon State U.	Corvallis	"

Details: Thermal neutrons, steady state, burner. Fuel-moderator: 20% enriched U as UO_2 dispersed homogeneously in polyethylene. Coolant: ambient air, natural convection. Reflector: graphite. Core structure: fuel-moderator discs 10 in. in diameter and of different thicknesses, four discs 1.5 cm thick, 3 discs .75 cm, and 2 discs .375 cm. Core and reflector in tank of H_2O serving as shielding. Control: safety and shim rods of polyethylene and UO_2; core divided into two halves connected by high-U^{235}-content polyethylene fuse, which melts in case of high flux level, causing lower half of core to fall 2 in.; negative temperature coefficient of reactivity. Power: negligible except for AGN-201-P-103 (20 Watts) and ACN-201-M-105 (5 Watts).

Code: 0313 16 31714 43 5931 711 83119 921 105
84677
85XX9

* Orig. Atoms for Peace Reactor, Rome & 2nd Geneva Conf.

No. 5 AGN-211- Reactor

Aerojet-General Nucleonics

References: Directory of Nuclear Reactors, III, 1960; Proc. Second U.N. Int. Conf., 10, pp. 368-74.

Originators: Staff members.

Status: AGN-211- (all 4 are identical)

No.	Owner	Location	Startup
-100	U. of Basel (at Brussels Intl. Expos. 1958)	Basel, Switzerland	1958
-101	Rice U.	Houston, Texas	1959
-102	U. of Oklahoma	Norman	1958
-103	U. of West Va.	Morgantown	1959

Details: Thermal neutrons, steady state, burner. Fuel-moderator: 20% enriched U as UO_2 dispersed homogeneously in polyethylene. Coolant: H_2O. Reflector: graphite. Fuel element: 12 in. long, approx. 3 in. square fuel-moderator section with 6 in. graphite reflector at each end. Element clad with polyethylene to prevent contact between fuel and coolant. Fuel elements surrounded by graphite reflector elements. Core in tank or pool of H_2O. Control: 2 Boral safety rods, 1 Cd and 1 Al coarse control rods, and 1 stainless-steel fine control rod. Power: 100 W(t). Specific power: 0.13 kW/kg. Power density: 5×10^{-3} kW/liter.

Code:									
	0313	16	31101	43	5931	711	84677	921	105
							81111		
							81112		

No. 6 Training, Research, and Isotope Production Reactor, General Atomic (TRIGA). Mark I and Mark II

General Atomic, Division of General Dynamics Corp.

References: Brochure, General Atomic, 1958; Directory of Nuclear Reactors, II, pp. 223-26 ff.; Programming and Utilization of Research Reactors, 2, pp. 91-115; NP-10188; NP-10714; Paper 57-AIF-3, 4th Ann. Conf. of AIF, 1957.

Originators: Staff members.

Status:

Owner	Location	Power, kW(t)	Startup (Dismantled)
	Mark I reactors		
Gen. Dynamics	La Jolla, Calif.	250	1958
Omaha Vet. Admin.	Omaha, Nebraska	18	1959
U. Arizona	Tucson	100	1958
U. Texas	Austin	250	1963
U. Minas Gerais	Belo Horizonte, Brazil	30	1960
U. Lovanium	Leopoldville, Congo Republic	250	1959
	Mark II reactors		
Columbia U.	New York City	250	being built
Gen. Dyn.-World Agricul. Fair	San Diego	50	1960 (1960)
U. Illinois	Urbana-Champaign	100	1960
Cornell U.	Ithaca, N.Y.	10	1963
Kansas State U.	Manhattan	10	1962
Ntl. Cmte. for Nucl. Research	Rome, Italy	100	1960
J.Stefan Nucl.Inst.	Ljubljana, Yugoslavia	250	being built
Musashi U.	Kawasaki City, Japan	100	1962
Rikkyo U.	Yokosuka City, Japan	100	1961
Korean AEC	Seoul, Korea	100	1962
Inst. Nucl. Res.	Dalat, Vietnam	250	1963
Inst. AE	Bandung, Indonesia	250	1964
U. Pavia	Italy	250	planned
Vienna Poly. Inst.	Vienna, Austria	100	1962
Inst. Tech.	Helsinki, Finland	100	1962
U. Mainz	Germany	100	being built

Details:

Mark I: Thermal neutrons, steady state (but can be pulsed), burner. For training and irradiation. Fuel-moderator: uranium, 20% U^{235}, homogeneously mixed with zirconium hydride. Coolant: H_2O. Additional moderator: H_2O. Reflector: graphite. Fuel-moderator elements clad in aluminum, in shape of cylindrical rods 28.4 in. long, 1.48 in. diameter. Elements arranged in circular lattice in right cylindrical core, about 14 in. in diameter, 14 in. high. Core surrounded by reflector set in tank or pool of coolant-moderator H_2O. Control: B_4C powder clad with Al shim-safety, safety, and regulating rods; samarium burnable poison discs at each end of fuel elements; prompt negative temperature coefficient of reactivity.

Mark II: Essentially similar to Mark I reactors.

Code:								
0313	17	31101	43	5981	711	84677	921	105
0323	13					81111		
						81164		

No. 7 TRIGA Mark III Reactor

General Atomic

Reference: GA-4339 Rev.

Originators: Staff members.

Status: Reactors being built at Univ. California (Berkeley) and Natl. Inst. Nuclear Energy (Salazar, Mexico).

Details: Thermal neutrons, steady state or pulsed, burner. Fuel-moderator: U (20 w/o U^{235})-ZrH_x -- 8.5% U, 89.9% Zr, 1.6% H. Fuel-moderator clad with stainless steel. Coolant: H_2O. Reflector: graphite, H_2O. Core arrangement: cylinder of about 80 cylindrical fuel-moderator elements (28.37 in. long, 1.47 in. diam.), 41 graphite dummy elements, and control rods in pool of H_2O. Reflector consists of graphite end sections on fuel elements and radial reflector of H_2O and of graphite in thermal-column position. Control: large prompt negative temperature coefficient, borated graphite rods for shim, safety, regulating, and transient control. Power: steady state, 1 MW(t); pulsed, 200 MW(t). Neutron flux (neutrons/cm^2/sec): steady state -- 3×10^{13}; pulsing: transient bursts up to 10^{17}; 6×10^{16} with 10 msec pulse width.

Code:									
	0313	17	31101	43	5981	711	84677	921	105
	0323	13					81111		

No. 8 TRIGA Mark F Reactor

General Atomic

Reference: Programming and Utilization of Research Reactors, 2, pp. 91-115.

Originators: Staff members.

Status:

Owner	Location	Power, kW(t)	Startup
Gen. Dynamics	La Jolla, Calif.	1500	1960
Northrop Corp.	Hawthorne, Calif.	100	1963
Diamond Ordnance Fuze Lab.* Rad. Facil. (US Army) (DORF)	National Bureau of Standards, Washington, D.C.	100	1961
Armed Forces Radio-biol. Res. Inst. (DASA) (AFRRI)	Bethesda, Md	100	1962

Details: Thermal neutrons, pulsing, burner. Built specifically for pulsing purposes. Differences from Mark I:

1. Graphite reflector eliminated.
2. Core contains more fuel elements, with higher heat capacity.
3. Larger H_2O tank for more versatility of irradiations.
4. Entire core is movable; Al support structure hung from carriage that moves on rails on reactor-room floor.

Mark F prototype can be pulsed to about 1500 MW(t). Pulsing parameters:

Reactivity insertion	2.2% $\delta k/k$
Max. power level	2000 MW
Prompt energy release	24 MW - sec
Min. pulse width	10 msec
Min. period	3 msec
Max. repetition rate	12 per hr

Shorter neutron lifetime in H_2O-reflected reactor allows smaller pulse width and higher max. power.

Code:								
0323	17	31101	43	5981	711	84677	921	105
	13					81111		
						81166		

*Now the Harry Diamond Laboratories

No. 9 TRIGA Variation

General Atomic

Reference: Nucleonics, 18, No. 12, p. 31, Dec. 1960.

Status: Proposal, 1960.

Originators: Staff members.

Details: Extrapolation of TRIGA technology for small reactor producing 25-50 kW(e) to deliver 35-70 hp for powering a bathyscaphe, research craft, or power buoy for the U.S. Navy.

Code:	0313	17	31101	43	5981	711	84677	921	105
		13					81111		

No. 10 Research Reactor, General Atomic REGA 10-30

General Atomic

Reference: Paper 57-AIF-3, 4th Annual Conf., AIF.

Originators: Staff members.

Status: Design, 1957.

Details: Core identical with TRIGA-Mark I core.

Code:	0313	17	31101	43	5981	711	84677	921	105
		13					81111		

No. 11 Isotope Production Reactor, General Atomic (IRGA)

General Atomic

Reference: Brochure, General Atomic; Paper 57-AIF-3, 4th Annual Conf., AIF.

Originators: Staff members.

Status: Design, 1957.

Details: Core identical with TRIGA-Mark I core.

Code:	0313	17	31101	43	5981	711	84677	921	105
		13					81111		

No. 12 Low-Power Research Reactor

North American Aviation, Inc.

Reference: NAA-SR-34.

Originators: A.S. Thompson and T. Fahrner.

Status: Design, 1959.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: graphite impregnated with enriched U as U_3O_8. Coolant: D_2O. Reflector: graphite. Core: hexagonal prism 4½ ft high. 37 Al-clad, helium-filled fuel elements; each contain hexagonally shaped fuel-moderator blocks. Alternatively, fuel-moderator blocks could be cylindrical or wedge-shaped. Coolant passes through concentric Al tubes located axially in each fuel element; inlet temp. 142°F, outlet 156°F. Control and safety rods in annular gap between core tank and reflector. Reflector, 30 in. thick graphite, surrounds core. Power: 200 kW(t).

Code:	0313	12	31102	4X	5931	711	8111X	921	105

No. 13 Siemens Unterrichts (Teaching) Reaktor (SUR)

Siemens-Schuckertwerke A.G.

Reference: At. Ind. Forum Memo, 8, No. 11, p. 14, Nov. 1961.

Orignator: Staff members.

Status: Construction proposed, 1961.

Details: Thermal neutrons, burner. Fuel-moderator: U_3O_8 suspended in polyethylene. Reflector: graphite. Shielding: Pb, H_2O. Power: 0.1, 1, or 10 W.

Code: 0313 16 3XXXX 4X 593X 711 8XXXX 92X 10X

No. 14 Homogeneous Subcritical Assembly

Aerojet-General Nucleonics

Reference: TID-7619, pp. 80-87.

Originators: Staff members.

Status: Two in operation, 1961 at The University of Texas and Pensacola Junior College, Florida.

Details: Thermal neutrons, subcritical assembly. Fuel-moderator: homogeneous mixture of polyethylene and 20%-enriched U as oxide. Coolant: ambient air. Reflector: graphite, polyethylene, or none. Neutron source: Pu-Be in core or an accelerator target source. Core: right cylinder. Purpose of assembly: to measure neutron fluxes.

Code: 0333 16 31714 43 5932 711 8XXXX 911 105
921

References

1. A.T. Biehl, D.C. Hetrick, and G.A. Bennett, A Table Model Reactor, Nucl. Sci. Technol. (Extracts from Reactor Sci. Technol., 2, Nos. 1-4, Apr.-Dec. 1952), TID-2503 (NAA-SR-Memo-352).

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Chapter 3. Power and Breeder Reactors

Most of the concepts described in this chapter are for high-temperature, gas-cooled reactors, some of which have been developed into large-scale plants. A few are liquid-cooled reactors for special purposes, such as the SNAP reactors. Typically, the reactor core is composed of fuel-moderator elements around which the coolant flows.

Graphite or beryllium oxide is the moderator for nearly all power reactors, with graphite being used for most. Beryllium metal was suggested early in the development of solid homogeneous reactors, but it has been little used. The advantages of graphite and those of beryllium oxide have been given by several authors. Faris[1] discussed graphite. Jaye et al.,[2] and Roberts,[3] gave the advantages of beryllium oxide. Desirable characteristics listed for graphite as a moderator in solid homogeneous reactors are: a negative temperature coefficient, which gives safety in low-temperature applications; desirable mechanical properties at high temperatures; comparatively simple decontamination; and simplicity of fabrication and recovery processes. Many advantages are claimed for beryllium oxide. Because it is a better moderator than graphite, conversion ratios are higher and cores are smaller with beryllium oxide than with graphite for the same fuel load and specific power. Higher temperatures for the gas coolant leaving the reactor are possible; thus more efficient steam cycles can be used and the temperatures might be high enough for efficient gas turbines. Beryllium oxide is compatible with carbon dioxide at high temperatures, so that this gas can be used instead of helium. With carbon dioxide there are fewer problems with gas-tightness in the system. The fairly high thermal conductivity of beryllium oxide makes it suitable for solid homogeneous fuels. It resists release of fission products from the fuel-moderator, and it may have a long lifetime without serious loss of essential properties. Jaye et al.[2] have suggested that a beryllium oxide spine might be used in a moderator element of graphite. One obvious major difference between the two moderators is the considerably higher cost of beryllium oxide.

Gas-cooled Reactors

Studies of the use of beryllium oxide in a high-temperature power reactor were begun by Farrington Daniels in 1944 at the Metallurgical Laboratory, University of Chicago. Calculations by Daniels and co-workers established the feasibility of such a reactor, and problems in its design were investigated at the Metallurgical Laboratory until 1946, when this work was transferred to a group at the Clinton Laboratories. An aim of the program was developing a high-temperature, helium-cooled, power reactor fueled with enriched uranium and moderated with beryllium oxide. A boiler for steam generation and a generator to produce electrical power were also included in the specifications for the reactor plant. Emphasis was put on a design that could be developed in a fairly short time.[4]

In November 1946, a preliminary design was proposed.[5] In this 40 MW(t) reactor, the fuel-moderator is in the form of enriched uranium oxide mixed with beryllium oxide to form cylinders, which are within axial holes in hexagonal prisms of beryllium oxide. These prisms form a cylindrical core. A reflector, a thorium breeder blanket, and the core vessel surround the core. The coolant, helium, flows into the bottom of the reactor, through the core, and out the top, whence it passes to a boiler.

At the suggestion of Wigner, a group studied a reactor utilizing beryllium metal instead of beryllium oxide. This study was published in February 1947.[6] The design is very similar to the earlier one with beryllium oxide. Plates of beryllium-uranium alloy are contained in a skeleton of beryllium metal. Fertile material, thorium metal, is in channels in the side and end reflectors of beryllium metal. The power is the same as for the oxide reactor, 40 MW(t). The results showed no outstanding differences between the moderators that would make the metal preferable. Also, the metal has such disadvantages as poor tensile and creep properties at high temperatures, and the knowledge of fabrication techniques for it was limited. Thus it was decided that use of the metal would give no advantage, and the development of a beryllium oxide pile was continued.

Investigations showed that reducing the volume of the pile would give a considerably smaller critical mass and would increase the conversion ratio. Also, tests on beryllium oxide indicated loss of thermal conductivity under irradiation, causing uncertainties in performance of the pile. It was thus decided to design a smaller, lower-power pile to give a better facility for study of such problems. In June 1947 the USAEC notified the Laboratory that it would not authorize design or construction of the power pile, but investigations on problems should continue at a lower priority.[4] In September of the same year, the Power Pile Division (the Clinton Laboratory group assigned to the project), issued a final report summarizing the status of the development.[4] The design advanced in this report is generally similar to that for the earlier design with beryllium oxide. The power is lower, 12-20 MW(t), and beryllium metal or graphite are considered as alternatives to beryllium oxide. Concepts for higher power were considered, as was a design for a horizontal reactor, intended to facilitate loading and unloading of elements. These concepts differed from the 12-20 MW(t) concept chiefly in dimensions, fuel loadings, and engineering.

A 1950 concept by Daniels, the Impregnated Graphite, Nitrogen-Cooled Reactor, was designed to use the standard equipment and materials available in 1950.[7] The author said that such a power pile could be built quickly and at low cost. It was intended to be a "one-shot" experimental reactor with expendable parts. The fuel-moderator is uranium oxide dispersed in graphite in the form of blocks that make up the core. Nitrogen was recommended as a coolant gas because it could be used to run standard turbines, but helium could be used. The reactor was designed as a breeder, with fully enriched uranium to be used for breeding U^{233} and slightly enriched uranium for breeding plutonium. For breeding U^{233}, thorium nitrate would be in the fuel-moderator or in the graphite reflector. Plutonium would be bred from the U^{238} in the fuel. The coolant flows from the top of the core to the bottom through channels in the graphite blocks and exits at 1100°F. The maximum power is 11.5 MW(t).

Another concept by the same author in 1956 was the Gas-Cycle Reactor,[8] in which the fuel-moderator consists of uranium carbide, which could be in different forms, such as rods or pellets. This reactor was also designed for high temperatures, with an exit coolant temperature of 1350°F. The power is 20 MW(t).

Siegel, in 1951, suggested a power breeder reactor in which the fuel-moderator is graphite impregnated with uranium dicarbide.[9] The coolant, helium or liquid bismuth, passes through axial coolant channels. Thorium fertile material is in either the core or a breeder blanket. The power is 1400 MW(t).

Investigations at the Studebaker-Packard Corporation, reported by Thompson in 1956,[10] resulted in a design for a gas-cooled reactor, with solid homogeneous fuel-moderator, intended for use with a gas turbine power plant. The fuel-moderator is a ceramic of uranium dioxide and graphite. Additional moderation is provided in the roughly spherical core by a central island of graphite or beryllium and a reflector of the same material. The fuel, with passages for flow of the helium coolant, is in the annulus between the island and the reflector. The power is 60 MW(t).

In a 1960 patent, applied for in 1956,[11] Fortescue and Lockett described a reactor with solid homogeneous fuel-moderator elements. The materials of the elements are not given, nor is the gas coolant. The fuel elements are surrounded by a reflector of graphite. The coolant gas passes downward between the reflector and the pressure shell of the reactor, then upward between core elements.

In Japan, work on "semi-homogeneous" reactors (which utilize solid homogeneous fuel-moderators) includes designs for two power reactors. For one design, a critical experiment has been operated.

Inoue et al.[12] in 1958 proposed that enriched uranium oxide, impervious graphite, and thorium oxide be incorporated into elements for a breeder reactor with a power of 600 MW(t). A blanket of thorium metal surrounds the core. Few details were given for the structure of this reactor, which is cooled by carbon dioxide. The outlet temperature of the coolant gas (700°C), according to the authors, would make possible use of the reactor with a gas or steam turbine.

The Semi-homogeneous Critical Experiment (SHE)[13-15] was reported in 1961 by Inoue and co-workers. The fuel-moderator is a uniform mixture of 20%-enriched UO_2 and graphite, formed into discs. The discs are placed, within a graphite tube, in sequence with discs of graphite to form rods. The rods, in hexagonal horizontal arrangement, are in two halves of the assembly. The halves are brought together for criticality. The experiment operates at room temperature. Discs of other compositions, e.g., thorium oxide for breeding, can be added to vary the core composition. Control rods are provided. The first loading was in 1961, and operation of the experiment was continuing in 1964.

The Semi-homogeneous Gas Cooled Breeder Reactor (SHR)[16] is very similar to the critical experiment, but it is designed for a power of 25 MW(t) and is cooled by helium. The fuel-moderator, UO_2 or UC_2 particles smaller than 6 μ evenly dispersed in graphite, is in the form of pellets. They are contained in an impervious graphite sheath. A blanket of thorium surrounds the core.

The Peach Bottom (Pennsylvania) High-temperature Gas-cooled Reactor (HTGR)[17-24] utilizes a design intended to have the advantages of high outlet temperatures for the gas coolant (1360°F or higher); high fuel burnup; high power density; and good neutron economy. The Peach Bottom reactor is intended as a prototype for reactors of this kind.

The preliminary design and development of the HTGR began early in 1957 at General Atomic Division, General Dynamics. The project is supported by High Temperature Reactor Development Associates, a group of 52 private utility companies. Research and development is being carried out by General Atomic under a contract with the USAEC. The Philadelphia Electric Company is owner and operator of the plant. Construction began in 1962. Fuel loading and start-up were scheduled for early 1965 but were delayed by a fire in the containment shell.[24] The reactor went critical on March 31, 1966. Power operation is now scheduled for late 1966. The reactor, designed as a converter, utilizes U^{235} as initial fuel and will use U^{233} after the breeding cycle has begun. Helium is the coolant, and thorium dicarbide is the fertile material. The cylindrical fuel-moderator element is of compound structure, with graphite as moderator, cladding, and fuel matrix containing the fuel and fertile material in homogeneous dispersion. The fuel-fertile material is a mixture of uranium and thorium carbides, as particles coated with pyrolytic carbon to control escape of fission products. There is an upper reflector, a fuel-bearing middle section, a bottom reflector, and an internal fission-product trap. The fuel elements are aligned vertically in a triangular pitch. The coolant gas flows upward in spaces between fuel elements. The power is 115 MW(t), 40 MW(e).

Advanced forms of the Peach Bottom reactor are under consideration[19,25] a 300- to 500-MW(e) reactor is the subject of a research and development program being carried out jointly by General Atomic and the Empire State Atomic Development Associate, Inc. (ESADA). With the same type of fuel elements as are in the Peach Bottom plant, the advanced design is for a reactor to produce steam for a 2400 psi, 1000°F superheat, 1000°F reheat steam cycle. Development and testing on extending fuel life are also in progress. A modified form of the Peach Bottom fuel is being evaluated for a 1000-MW(e) reactor plant, the Thermal, Advanced Reactor, Gas-cooled, Exploiting Thorium (TARGET).[19,25] The program is being carried out by General Atomic under contract to the USAEC. This concept is being considered for possible construction in the 1970's. The reactor would have about the same surface temperatures as in the Peach Bottom prototype, but a more efficient steam cycle would be used, to give a steam pressure of 3500 lb psi, superheat to 1050°F, and two reheats, each to 1000°F. In the most highly developed design,[25] the fuel, uranium carbide, is a bed of particles within a graphite container. Thus it is not a solid homogeneous reactor. In a suggested alternative, features are the same except that the fuel-moderator-fertile material consists of microscopic particles (1 μ or less in diameter) dispersed in a graphite matrix. Still another possibility being evaluated is operation with recycled U^{233} as fuel and thorium fertile material in the fuel-moderator elements to achieve core breeding.

The OECD High Temperature Gas-Cooled Reactor Project (DRAGON),[26-28] had its origin in about 1956, when a team of the United Kingdom Atomic Energy Authority, under Fortescue, began preliminary studies. The first report was the 1958 paper by Shepherd et al.[26] In that year, a committee was set up under the auspices of the OECD (Organization for Economic Cooperation and Development) to study collaboration among members of the organization, with a high-temperature gas-cooled reactor chosen for study. In 1959 an agreement was concluded by Austria, Denmark, Euratom, Norway, Sweden, Switzerland, and the United Kingdom for work on this type of reactor. Development of the reactor experiment is carried out at Winfrith Heath, England, by staff members from the parent organizations. The objectives of this reactor development are those of others utilizing solid homogeneous fuel, including the attainment of exit-gas temperatures (1382°F) higher than possible before.

The core has two zones, inner and outer. It is so divided because it is difficult to test, in a small core that has much higher neutron leakage, units comparable with those of a large reactor. In the inner zone the fuel, uranium carbide, is diluted with thorium carbide; in the outer it is diluted with zirconium carbide. The fuel-moderator is in the form of particles, which are pressed into compacts that are inserted into graphite fuel tubes. Seven tubes make up a fuel element. The core, which is made up of 37 of these fuel elements in a triangular lattice, is surrounded by a graphite reflector. The coolant gas, helium, passes upward through the core. The power is 20 MW(t).

The reactor became critical in August 1964.

The Experimental Beryllium Oxide Reactor (EBOR) is the land-based prototype for the Maritime Gas-cooled Reactor (MGCR), which is to be of higher power, but otherwise based on EBOR.[29-31] The purpose is to demonstrate the technology of beryllium oxide as moderator for gas-cooled reactors to be used in central station and marine power plants. EBOR, which was designed by the staff of General Atomics, is at the National Reactor Testing Station in Idaho. The fuel-moderator consists of compacts of highly enriched uranium dioxide and beryllium oxide, which are contained in Hastelloy-X fuel pins. The pins, which are arranged cylindrically, are surrounded by blocks of beryllium oxide. The coolant gas, helium, flows into and out of the reactor through two concentric passages at the side of the reactor. The exit temperature is 1300°F. The design of EBOR is for 10 MW(t).

The Ultra High Temperature Reactor Experiment (UHTREX),[32-37] formerly termed TURRET, is being developed at Los Alamos Scientific Laboratory, with completion scheduled for 1966. The design is for a coolant exit temperature (2400°F) appreciably higher than any previously used. The design (as TURRET) was originally suggested for electrical generation with a high-temperature gas turbine, but it was later changed to an experimental prototype for a process-heat reactor operating at an unusually high temperature for the exit gas. One purpose was to investigate operation at high temperature, e.g., the behavior of unclad porous graphite fuel elements at up to 3000°F. The fuel-moderator consists of hollow graphite cylinders containing uranium carbide as microspheres coated with pyrolytic carbon, which delays escape of fission products. The fuel-moderator elements are arranged in rows within channels in the cylindrical graphite core. Mechanical rotation of the core over loading rams permits reloading the core during operation at full power. Helium coolant enters the core at the bottom and, after the flow is divided into fuel channels, leaves at the top for discharge. The power for this experiment is 3 MW(t).

The use of beryllium oxide as moderator in a high-temperature pebble-bed reactor (the HTGC, High Temperature Gas Cooled Reactor) is being investigated by workers of the Australian Atomic Energy Commission.[38-41] Beryllium oxide is the matrix for a solid ceramic fuel-moderator-fertile material, in which plutonium and thorium oxides are dispersed. It was designed to incorporate the advantages of this moderator, which have been previously discussed in this chapter. The pebble-bed concept was chosen because it permits high burnup (by recirculation of the pebbles) and because no neutron absorbers need be included in the fuel-moderator. The study of beryllium-moderated, high-temperature, gas-cooled reactors was begun by the Australian AEC in 1960, and the concept is being assessed now for plutonium fuel; U^{233} and U^{235} fuels will be considered later. Currently, the design is for a cylindrical core vessel containing the pebbles, which are slowly added at the top and removed at the bottom. The carbon dioxide coolant flows up between the core vessel and the pressure vessel to the top then downward through the core, leaving the bottom at 700-800°C. The power density is 10-20 kW/liter.

A brief description of a helium-cooled solid homogeneous reactor was published in 1963 by Sternglass *et al.*[42] The reactor is part of a system designed to utilize MHD (magnetohydrodynamic) power generation. In the 264-MW(e) reactor, the core is a graphite cylinder, in which fuel-moderator elements, of uranium carbide uniformly distributed in graphite, are contained within vertical channels. The homogeneous fuel-moderator, which is unclad, was chosen because the high temperatures required by the MHD unit would prevent the use of clad fuel.

Other Reactors for Electrical Power

Although most solid homogeneous reactors for electrical power are gas-cooled, some concepts have been advanced for reactors cooled by other fluids.

In a 1960 United Kingdom Patent,[43] applied for in 1956, the fuel, moderator, and fertile material are combined as a mixture to form fuel rods. A typical mixture would be $U^{233}O_2$, BeO, and ThO_2. Boiling water is the coolant, but because only enough of it to cool the fuel elements is used, it does not act as moderator.

In an alternative to the gas-cooled Japanese Semi-homogeneous Gas Cooled Breeder Reactor described earlier,[16] Inoue et al. suggested that liquid bismuth, rather than helium, be used as coolant.

A low-power [about 100 kW(e)] reactor is being developed at the Martin Nuclear Division of the Martin-Marietta Company for a direct-conversion system, Terrestial Unattended Reactor Power System (TURPS).[44,45] The system is for a self-regulating constant heat source at about 1000°F. In an early report by Murphy, the coolant would be phosphorus sesquisulfide, P_4S_3, and alternatives were given for fuel and moderator: uranium dioxide as fuel and water as moderator, or a fuel-moderator of $ZrUH_x$. Boiling P_4S_3 was chosen as coolant because this compound has low neutron absorption and resists radiation; little is known, however, of its flow and heat-transfer characteristics. The criteria were for a low-cost, portable, system of medium size that would be reliable, be simple to operate, and have low capital and operating costs. A plant with a net power of 1.3 MW(e) was originally proposed. A few details of a more recent design have been published. The coolant is the same, P_4S_3. The fuel-moderator element has two segments: one is an alloy of uranium and zirconium hydride; the other is zirconium hydride alone. In the configuration developed, hydrogen can migrate from the unfueled segment to the fueled segment to act as moderator. Control could be by reduction of heat load in the unfueled section, which would reduce hydrogen migration and thus reduce activity. The coolant circulates by natural convection in a boiling-condensing cycle. A contract for investigation of the concept has been awarded by the United States Air Force.

Reactors for Aircraft and Space Vehicles

Several reactors designed for use in aircraft, as power sources for space propulsion, or as auxiliary power for space vehicles utilize solid homogeneous fuel. Some of the concepts have been proposed for aircraft, while the Tory, SNAP, Kiwi, Phoebus, and NERVA reactors have been prominent in the development of space vehicles.

In 1959 patents, filed in 1949,[46,47] Grebe described a reactor for propulsion in which the porous fuel-moderator element is in the form of porous segments of fuel-moderator, separated by coolant passages. Because of the structure of this core, the name "Cabbage Head" has been applied to it.[48] The segments converge to close at one end and are joined to adjacent segments at the other end. The segments bow outwardly, the bowing increasing as the distance from the center increases, so that a roughly spherical or pear-shaped core is formed. Gas flows into passages, through the porous segments, out coolant channels, and through a jet nozzle. The patent claims that the design permits close contact of the coolant with fissionable material and the flow of a large volume of gas without excessive pressure loss. In a modification, the use of different moderator matrix materials in different zones was proposed. A refractory moderator, such as graphite, would be used in the hot zone near the nozzle, and a less-refractory moderator, of better nuclear properties, would be used in the cooler inlet zone.

An early (1952) concept by staff members at the Bendix Corporation is for a low-power water-cooled reactor in which U^{235} is distributed in zirconium hydride.[49]

A design for a high-temperature reactor utilizing a ceramic of highly enriched uranium dioxide in beryllium oxide with yttrium hydride (which has a much higher hydrogen content than zirconium hydride) as additional moderator was proposed by Leeth in 1958.[50] The UO_2-BeO and the yttrium hydride are arranged homogeneously in concentric polygons. The coolant, air, has an exit temperature of between 1800°F and 2000°F.

In 1960, Levengood, Dissler, and Kalinowski described a reactor, the PWAR-II (Pratt & Whitney Aircraft Reactor - II), in which the fuel-moderator is uranium dioxide in beryllium oxide.[51] Lithium coolant flows around the hexagonal containers for the fuel-moderator pins. Beryllium oxide is also the reflector. Designs for 510 MW(t) and 200 MW(t) reactors were given.

Newgard and Leval described in 1960 a design for rocket propulsion in which uranium-loaded graphite plates form a core through which hydrogen propellant flows.[52] The reflector is made up of beryllium oxide blocks, and control is by movement of control plates in the reflector.

As part of the PLUTO program to develop a reactor for a nuclear ramjet engine, two solid homogeneous reactors, Tory IIA-1 and Tory IIC, were developed and successfully tested.[53,54] The reactors are similar in design. In both, the cylindrical core consists of bundles of many fuel-moderator elements (100,000 for Tory IIA-1 and several hundred thousand for Tory IIC) of highly enriched uranium dioxide homogeneously mixed with beryllium oxide. A graphite reflector, with rotating cylinders containing poison rods, is used for part of the control.

Cooling is by single pass, straight-through flow of air from inlet, through the reactor, and out the exhaust.

Tory IIA-1 first operated on May 14, 1961. Full power, 155 MW(t), operation took place later in 1961. The operation of Tory IIC (May 20, 1964) at full power was also successful. In that year, emphasis on the project was reduced because of the decision that the work by the USAEC was essentially completed. The reactors were inactivated.

Work on the SNAP (Systems for Nuclear Auxiliary Power)[55-60] reactors originated in the need in advanced space vehicles for compact, long-lived, and high-power nuclear sources of electrical power. Atomics International, a division of North American Aviation, Inc., began investigating the problem in 1953, and by 1955 the type of reactor was chosen. The bases used in the choice included minimum weight and suitable operating temperature. In the general reactor design, the fuel is U^{235}, moderated with zirconium hydride. The coolant is a liquid metal, which transfers heat to a mercury Rankine cycle or a direct-conversion thermoelectric unit for power conversion. In 1956, the USAEC established the SNAP program.

A homogeneous thermal reactor with highly enriched fuel and a reflector was chosen for several reasons. The homogeneous core with enriched fuel permits small size and efficient use of the fuel. A thermal reactor is preferable to a fast reactor because of the lower uranium inventory and less-complex control problems. Adding a reflector (beryllium) reduces the minimum critical mass of uranium in the core. The use of the zirconium hydride as moderator also gives a smaller core size. To restrain thermal dissociation of this hydride, it is clad with a material that is a barrier to diffusion of hydrogen. Control of the reactor is accomplished by varying the thickness or effectiveness of the reflector, or by removing it to make the reactor subcritical.

The general design was used in a series of reactors: SNAP Experimental Reactor (SER); SNAP-2 Development Reactor (SDR or S2DR); SNAP-2; SNAP-4; SNAP-8; SNAP-10; and SNAP-10A. The first two were preliminary developments for the SNAP-2, SNAP-8, SNAP-10, and SNAP-10A reactors. Both operated for test periods. SNAP-2 was intended as a low-power [3 kW(e)] reactor for space application for use with a mercury Rankine power conversion cycle; it was a predecessor of SNAP-8. This project was redirected in 1963, although the Rankine cycle was used in later developments. SNAP-8 is a joint project between the USAEC and the National Aeronautics and Space Administration for developing a reactor of 600 kW(t), 35-50 kW(e), for space applications with a mercury Rankine cycle.

SNAP-10 and SNAP-10A are direct-conversion systems. In SNAP-10, the heat is converted by conduction from the core to a thermoelectric system. In SNAP-10A, the heat from a liquid metal coolant is converted by means of a thermoelectric system.

The SER, SNAP-2, SNAP-8, SNAP-10, and SNAP-10A differ chiefly in power, dimensions, heat conversion, and some other details, but they are basically of the general type that has been discussed.

The basic core configuration for the SNAP reactors consists of the fuel-moderator rods, of uranium and zirconium hydride in homogeneous mixture, arranged in a triangular pattern to make up a hexagonal core. Rods are clad with steel or Hastelloy: stainless steel for the SER and Hastelloy-N for the later reactors. The beryllium reflector is in segments and contains control drums, which can be rotated to change neutron leakage. For scram, three beryllium safety plates fall away. The NaK coolant flows axially among the fuel elements. In SNAP-10A the spaces between the angles of the hexagonal core and the cylindrical core vessel are filled with beryllium metal.

The SER,[55,56] 50 kW(t), operated from September 1959 until the end of the test program, in November 1960. During more than a third of the operating time the outlet temperature of the coolant was 1200°F.

SNAP-2 Development Reactor (S2DR, SDR)[55-57] was a second-generation design, which was to approach the requirements of a space reactor. It operated successfully from April 1961 until December 1962.

A prototype of SNAP-8,[55-57] with a higher power, 600 kW(t), was undergoing test operation in 1965.

The design for SNAP-10[58,59] incorporates cooling by direct conduction of heat from the core to thermoelectric devices on the reactor face. Thus a liquid-coolant system, with pumps, etc., is eliminated. The core consists of stacks of uranium-zirconium hydride plates, with beryllium plates alternating to increase heat transfer. This reactor was to be controlled by a strong negative temperature coefficient; for startup or shutdown, halves of the reactor are joined or separated. This concept originated from a requirement of the U.S. Air Force for a device to produce 200-250 Watts (electrical). In 1958-59 Atomics International developed this concept, with work continuing until 1960. Some reasons for redirecting the project were: the temperature range desired (1100°F) was too high for the thermoelectric material then used (lead telluride); and there was little prospect of the solid-fin-type heat-rejection radiator being developed to increase the power much beyond 300 Watts. With a liquid-metal coolant, as used in the SNAP-2 reactor, the higher powers are possible.

SNAP-10A[56,57,59] is essentially the SNAP-2 concept with lower power, 34 kW(t), and lower outlet temperature for the coolant, 1010°F instead of 1200°F. The bulk of the reflector is outside the core, and it can be completely removed without affecting the coolant system. The direct-conversion system utilizes silicon-germanium alloy as the thermoelectric material. The conversion system has an overall efficiency of 1.63%. SNAP-10A is the first nuclear reactor to be orbited in space. Work on this reactor was begun in 1961 and performance testing was begun in December 1962. A SNAP-10A reactor (FS-3) operated in a ground test for 10,000 hours from January 1965 to March 1966. On April 3, 1965, another SNAP-10A (FS-4) was orbited in a spacecraft. It operated until May 16, 1965, when the system shut down. The cause given as most probable for the shutdown was a failure in the electrical system of the satellite carrying the SNAP-10A, rather than in the reactor itself.

SNAP-4 differs from the other SNAP reactors in purpose and in the coolant used. A project on this reactor began in 1959, with the aim of developing a reactor and turboelectric system that would be very compact and capable of long unattended operation under water or in a remote land location.[61] The power level is 1-4 MW(e). The design is very similar to that of the SNAP-2 and SNAP-8. The fuel-moderator is hydrided zirconium-uranium alloy as rods in bundles that make up a vertical hexagonal core.[62] Unlike the other SNAP reactors, SNAP-4 utilizes boiling water as coolant. Water boils in the core and the steam goes to a turbine. Advantages claimed for the design are compactness; feasibility (built with existing technology); low cost; safety; no maintenance during core life; versatility in application; and automatic, unattended operation.[62]

A prominent application of solid homogeneous fuels in space propulsion is in some of the Kiwi reactors and reactors developed from them.

Under Project Rover, which began at the Los Alamos Scientific Laboratory in 1955,[63] development of solid-fueled reactors as potential sources of flight power was begun. An aim was to supply basic reactor design for a flight engine. The reactors would produce high temperatures and would pose no unexpected problems in startup, stable operation, or shutdown.[64]

The first integral reactor test was of Kiwi-A.[64-66] This reactor was successfully tested on July 1, 1959. A power of 70MW(t) and an exit-gas temperature of 3200°R were achieved. The fuel element is made up of graphite plates containing enriched uranium oxide. These plates, stacked within cylindrical boxes, are in an annular zone around a central island containing heavy water and control rods. A graphite flow-separator cylinder, a graphite outer reflector, and a double-walled, water-cooled aluminum pressure vessel surround the fuel zone. The gaseous hydrogen coolant enters the reactor at the upper end of the reflector, passes down into the core, up between the fuel plates, into an exhaust, and out the nozzle.

After the testing of Kiwi-A, several designs were developed and tested, with the aim of producing a 1000-MW(t) reactor. Liquid, instead of gaseous, hydrogen was the coolant used in some later developments, and some changes in core design were introduced. Even the early Kiwi tests showed the potential of such reactors and stimulated interest in developing a flight engine.[63]

In mid-1961 the NERVA (Nuclear Energy for Rocket Vehicle Application) program was begun under the direction of the NASA-AEC Space Nuclear Propulsion Office. The Aerojet-General Corporation was designated as the prime contractor, with Westinghouse Electric Corporation as the principal subcontractor for development of the reactor, shielding, and reactor controls.[63]

In the development of the Kiwi reactors, Kiwi-B1 was designed for liquid hydrogen, but the first Kiwi-B reactor that was operated (Kiwi-B-1A) utilized gaseous hydrogen.[65] It was tested on December 7, 1961, attaining a power of 330 MW(t). The test of Kiwi-B-1B (1962) was the first test in which the reactor was started and brought to high power with liquid hydrogen as coolant.[67] Testing continued, with Kiwi-B-4E reaching 1000 MW(t). With Kiwi-B-4E, improved fuel elements were utilized. The uranium is in the form of extremely small beads of uranium carbide coated with pyrolytic carbon. The power for the test of this reactor (August and September 1964) was nearly 900 MW(t), and the temperature of the coolant leaving the fuel elements was about 4000°R.[65] Among the developments in the Kiwi-B series was the demonstration that the reactors could be controlled by drums in the reflector.[68]

The next step in the development was a reactor designed and built by Westinghouse Electric and tested by Aerojet-General Nucleonics. This design was based on Kiwi-B-4A.

Two of the chief problems in designing a reactor for flight were in developing a structure and fuel elements that would withstand such conditions as extremely high temperatures, vibration, and corrosion. The reactor, NRX (NERVA Reactor Experiment), was given a non-nuclear test (NRX A-1) in the spring of 1964 to test stability and structure. NRX-A2, a nuclear test, was successfully carried out in September and October, 1964.[63]

In the NRX-A reactor, the fuel element is enriched uranium carbide, coated with pyrolytic carbon, uniformly dispersed in graphite. The fuel-moderator elements contain passages for flow of the coolant (hydrogen), and they are grouped in clusters. There is an inner reflector of graphite and an outer one of beryllium. Control is partly by 12 cylindrical beryllium drums, within the outer reflector, that rotate to turn a poison strip to or away from the core. There are also control rods in the core and the outer reflector.[69]

Concurrent with the NERVA developments has been the Phoebus program at LASL.[65,68] The Phoebus reactors are advanced versions, with many improvements, of Kiwi reactors. They are large, high-power graphite reactors intended to provide long operating times, high specific impulse, and restart capability. Clustering of engines for propulsion is part of this program. Phoebus-1 operated in July, 1965 at its design power, 1000 MW(t).[70,71] Performance parameters of the Phoebus-2 design illustrate some of the conditions sought: a power of 5000 MW(t), a hydrogen flow of 285 pounds per second, and an exit-gas temperature (from the fuel elements) of 4500°R. Above 5000°R, graphite-based reactors, with solid-to-gas heat exchange, will probably not be feasible.[65]

DATA SHEETS

POWER AND BREEDER REACTORS

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No. 1 First Daniels Experimental Power Pile

Clinton Laboratories

Reference: MonN-188.

Originators: Farrington Daniels suggested pile; development completed by Power Pile Group (later Power Pile Division) at Clinton Laboratories.

Status: Preliminary design, 1946.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator: UO_2-BeO containing 2% of 50% enriched uranium. Coolant: helium. Fertile material: Th metal. Reflector: BeO and graphite. Core structure: elements of UO_2-BeO in form of hollow cylinders, 1½ in. OD, 1 in. ID, 4½ in. long. Cylinders within 2-in. axial holes in hexagonal BeO prisms, 3 in. across flats, 4½ in. long. BeO prisms form cylindrical core, 6 ft diam., 5½ ft high. 517 channels in core: 504 for fuel elements, 13 for safety and control rods. Helium flows into bottom of reactor (at 500°F), through annular spaces between UO_2-BeO cylinders and BeO prisms, and out top (at 1400°F) to go to steam boiler. Core surrounded by inner reflector, breeder blanket, outer reflector, stainless-steel liner, insulation, and carbon-steel pressure shell. Control: 6 safety rods and 7 control rods; 6 control rods in circle 15 in. from center of core and one at center. Power (max.): 40 MW(t); 10 MW(e).

Code: 0312 15 31716 43 5932 726 8111X 941 105

No. 2 Second Daniels Experimental Power Pile

Clinton Laboratories

Reference: M-4157.

Originators: Farrington Daniels suggested original pile; E.P. Wigner proposed Be metal pile; development by Power Pile Group (later Power Pile Division) at Clinton Laboratories.

Status: Design study, 1947.

Details: Thermal neutrons, steady state, converter. Fuel-moderator: U^{235} in U-Be alloy. Coolant: helium. Reflector: Be. Fertile material Th. Fuel elements: short hexagonal prisms made of plates of U-Be alloy, 3/16 in. thick. Elements 8.5 in. long, 7.9 in. in diameter. Core: hexagonal, horizontal cylinder with Be skeleton incorporating 37 fuel channels parallel to the axis, 4.60 ft diameter, 4.25 ft long. Helium at 10 atm. flows parallel to fuel elements; inlet temperature 500°F, outlet temperature 1400°F. Containment: austenitic-steel inner support for Be structure; low-alloy-steel outer support, which also acts as pressure shell. Fertile material contained in channels in side reflector and possibly also in end reflector incorporated into ends of fuel elements. Maximum conversion ratio: 0.9. Control: 8 vertical rods. Power: 40 MW(t).

Code: 0311 15 31716 44 5922 726 8111X 941 105

No. 3. Third Daniels Experimental Power Pile

Clinton Laboratories

Reference: MonN-383.

Originators: Farrington Daniels suggested original pile; development by members of Power Pile Group (later Power Pile Division) at Clinton Laboratories.

Status: Design; abandoned, 1947.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator: 30%-enriched uranium or uranium compound dispersed in matrix of BeO, Be, or graphite. Coolant: helium. Fertile material: ThO_2. Reflector: BeO. Fuel-moderator elements fit (with some clearance) within 2-in. axial holes in hexagonal BeO prisms, 3 in. across flats and 6 in. long. These prisms may, instead of the fuel-moderator elements, contain BeO alone or fertile material, depending upon whether they are to be fuel elements, reflector elements, or fertile elements. Prisms form 517 vertical channels, 2 in. diam., 76 in. long. Core is cylinder 72 in. diameter and 76 in. long. Cylinder retained by shell and supported at bottom by plate. Cylindrical steel pressure shell (8 ft OD, 1½ in. thick, 22 ft 6 in. long) surrounds this assembly. Helium flows, at 10 atm., into pressure shell at top at 500°F, flows downward through annulus inside shell, passes upward through channels in core, to a plenum at the top inside the pressure shell, leaving the reactor at 1400°F. It is then passed to boilers. Control: 1 control rod at core center; 6 shim rods in ring around center; 6 safety rods in outermost ring. Power: 12-20 MW(t); 2.4-4 MW(e). Designs for 60 and 250 MW(t) possible, with different dimensions, critical loadings, etc.; designs essentially similar to this concept. Horizontal reactor structure, similar in most respects to this concept, proposed to facilitate loading and unloading fuel elements.

Code:	0312	12	31716	43	5912	726	811X	941	105
		15			59X2				

No. 4 Impregnated Graphite, Nitrogen-cooled Reactor

Reference: ANL-4448 (AECD-4095).

Originator: Farrington Daniels.

Status: Design, 1950.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator: UO_2 dispersed in boron-free graphite; graphite impregnated with $UO_2(NO_3)_2$ and heated to form uranium oxide; fully enriched in U^{235} if U^{233} is to be bred, slightly enriched if Pu is to be bred. 10-15 kg U^{235} required. Coolant: N_2 or helium. Reflector: 20 tons of graphite 1½ ft thick, at sides, top, and bottom. Fertile material: for U^{233}, $Th(NO_3)_4$ in moderator or reflector; for Pu, U^{238} in fuel. Core: 370 impregnated graphite moderator blocks 4 in. x 4 in. x 45 in. Coolant, at 5 atm., flows into top of core, down through vertical channels in moderator blocks, and out bottom to turbine for power production. Coolant inlet temperature, 558°F; outlet temperature, 1100°F. High temperature allows both power production and breeding. Arrangement: impregnated graphite blocks rest on vertical slabs of BeO. Reactor is 6 ft diam., 5½ ft high. Containment: stainless-steel shell 9 ft diam. and 12 ft high. Control: boron-steel control rods in holes lined with BeO. Power: 11.5 MW(t); 1.9 MW(e) (designed for use with gas turbine available at time of proposal).

Code:	0312	12	31718	44	5931	736	81111	941	105
			31716	42		723			

No. 5 Gas-Cycle Reactor

Reference: Nucleonics, 14, No. 3, pp. 34-41, March 1956.

Originator: Farrington Daniels.

Status: Conceptual design, March 1956.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: 10% enriched U contained in graphite as uranium carbide. Fuel-moderator could be in form of rods, pellets in cartridges, or UO_2 and graphite powders could be mixed in cylinders and heated, producing uranium carbide. Coolant: helium. Pressure shell contains coolant at 225 psi. Coolant enters at 743°F and leaves at 1350°F. Control: regulating rods containing B or B_4C in Mo tubes. Power: 20 MW(t).

Code:	0313	12	31716	43	5941	711	81111	9XX	105

No. 6 High-temperature Bismuth-cooled Power Breeder

North American Aviation, Inc.

Reference: Sidney Siegel, unpublished report, Aug. 1951.

Originator: Sidney Siegel.

Status: Proposed design, 1951.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator: graphite impregnated with $U^{233}C_2$. Coolant: helium or liquid Bi. Fertile material: Th or compound in core or breeding blanket. Reflector: graphite. Fuel-moderator either solid graphite core (with axial coolant channels) impregnated homogeneously with fuel, or impregnated graphite tubes suspended in coolant channels. Th fertile material: could be impregnated in graphite, in a slurry of thorium deuteroxide in D_2O or as thorium fluoride pellets in graphite matrix. Core: cylinder, 3.4-6.8 ft diam., 3.4-6.8 ft high. Reflector around core. Coolant flows through axial coolant channels. Power: 1400 MW(t); 470-700 MW(e).

Code:	0312	12	31105	45	5942	7X6	8XXXX	941	105
			31716						

No. 7 Solid Homogeneous Reactor for Open-Cycle Gas Turbine

Studebaker-Packard Corporation

Reference: AECU-3559.

Originators: A.S. Thompson, staff members.

Status: Preliminary design, 1956.

Details: Thermal neutrons, steady state, converter. Fuel-moderator: compacted ceramic of UC_2 and graphite; 10% enrichment if Be is used for additional moderator; 20% if graphite. Additional moderator: Be or graphite. Coolant: helium. Inlet temp., 1240°F; outlet, 1740°F. Pressure: 20 atm. Reflector: graphite or Be. Core structure: central island (40 in. OD) of graphite or Be and reflector (100 in. OD) of same material, in roughly spherical reactor; fuel in annulus between island and reflector. Coolant flow through passages in fuel-moderator. Containment: horizontal Inconel pressure shell, 2 in. thick, 10 ft diam. Power: 60 MW(t).

Code:	0311	11	31716	43	5942	724	8XXXX	921	105
		15							

No. 8 Solid Homogeneous Type Gas Cooled Nuclear Power Reactor

UKAEA

Reference: United Kingdom Patent 850,014.

Originators: Peter Fortescue and G.E. Lockett.

Status: Patent granted, 1960; filed in 1956.

Details: Thermal neutrons, steady state, burner. Designed to operate at high temperature for power production. Fuel: not specified; fuel elements solid homogeneous, with moderator and fuel combined. Coolant: gas. Reflector: graphite. Each of 61 fuel elements contained on a spike projecting from perforated platform structure resting on brackets within cylindrical stainless-steel pressure shell. Fuel elements surrounded by ring of wedge-section graphite reflector blocks, which form barrier for the gas coolant. Another wedge of blocks surrounds this. Core and reflector assembly: elongated core elements pivotally supported at lower end and drawn together at the upper end by radial pressure differential of coolant. Coolant passes downward between reflector and pressure shell and upward between core elements. Coolant path determined by gas seals between core reflector, which are also maintained by pressure differential.

Code: 0313 1X 317XX 4X 5XXX 711 8XXXX 921 105

No. 9 Semi-homogeneous High-temperature Gas-cooled Breeder Reactor

Japan Atomic Energy Research Institute

Reference: AEC-tr-3620.

Originators: K. Inoue *et al.*

Status: Preliminary design, 1958.

Details: Thermal neutrons, steady state, breeder. Fuel: Initially 20% enriched U^{235} in UO_2; after breeding begun, U^{233} used instead of U^{235}. Moderator: graphite, coated with silicon carbide to make it impervious. Coolant: CO_2. Fertile material: ThO_2, Th metal. Fuel, moderator, and ThO_2 incorporated into elements that can be either rod- or slab-shaped and arranged either vertically or horizontally. Th metal in blanket of graphite around core. Coolant inlet temperature: 300°C; outlet, 700°C. Power: 600 MW(t).

Code: 0312 12 31717 43 5932 726 8XXXX 941 105

No. 10 Semi-homogeneous Critical Experiment (SHE)

Japan Atomic Energy Research Institute

References: Trans. Am. Nucl. Soc., 4, No. 1, pp. 56-7, June 1961; JAERI-1032; JAERI-4014.

Originator: K. Inoue et al.

Status: First loading completed January 1961; in operation, 1964.

Details: Thermal neutrons, steady state, burner or breeder. Critical assembly. Fuel-moderator: UO_2(20% U^{235}) - graphite. Coolant: air, at ambient (room) temperature. Reflector: graphite. Fertile material, if any: ThO_2. Fuel-moderator homogeneous mixture of UO_2 and graphite in cold-pressed discs 4.5 cm diam. and 1 cm thick, with graphite-to-UO_2 weight ratio of 10 to 1. Graphite tube contains, in sequence for its 120 cm length: fuel disc, 1-cm graphite disc, and 0.5-cm graphite disc. Rods arranged in hexagonal, horizontal array in two halves of assembly, which come together to achieve criticality. To vary core composition, discs of ThO_2, graphite mixtures, and pure graphite can be added. Control: rods containing B or Cd; 1 control and 3 safety rods in each half; large negative temperature coefficient.

Code:	0313	12	31714	43	5931	711	81211	921	105
	0312					726	81212		
							84677		

No. 11 Semi-homogeneous Gas Cooled Breeder Reactor (SHR)

Japan Atomic Energy Research Institute

References: IAEA Symp. on Power Reactor Experiments, I, pp. 149-54.

Originator: K. Inoue et al.

Status: Design, 1961.

Details: Thermal neutrons, steady state, breeder. Fuel: initially U^{235}, ultimately U^{233}. Coolant: helium. Moderator: graphite. Reflector: graphite. Fertile material: thorium. Fuel-moderator of UO_2 or UC_2 particles, smaller than 6 μ, uniformly dispersed in graphite as ring-shaped pellets. Pellets contained in impervious graphite sheath. Thorium blanket surrounds core. Graphite reflector between core and blanket. Power: 25 MW(t).

Code:	0312	12	31716	43	5931	726	8XXXX	941	105
				45	5941				

No. 12 Peach Bottom High-temperature Gas-cooled Reactor (HTGR)

Philadelphia Electric Co.

References: Proc. Third U.N. Conf., 5, pp. 101-115; J. Franklin Inst., Monograph No. 7, pp. 27-40, 1960; Nucl. Sci. Eng., 20, No. 2, pp. 201-218, Oct. 1964; At. Energy Rev., 1, No. 3, pp. 119-139, 1963; TID-7662, pp. 299-316; Nuclear Congress, N.Y., 1964, Preprint No. 91; Nuclear News, 8, No. 3, p. 9, March 1965; Atomics, Nov.-Dec., 1964, pp. 10-11.

Originators: General Atomic staff members.

Status: Critical, March 1966, power operation scheduled for late 1966.

Details: Thermal neutrons, steady state, converter. Fuel: initially 93.5% enriched U^{235}; U^{233} after breeding cycle has operated. Coolant: helium. Moderator: graphite. Reflector: graphite. Fertile material: ThC_2. Solid homogeneous fuel element, of which graphite is moderator, cladding, fuel matrix, and core structure, consists of upper reflector, fuel-bearing middle section, bottom reflector section, and internal fission-product trap. U^{235} and thorium dicarbides (coated with pyrolytic carbon) dispersed in graphite in annular fuel compacts 1.5 in. long, 2.75 in. in diameter, which are stacked on cylindrical spine in fuel element. Overall length of fuel assembly: 7.5 ft. Core structure: 804 cylindrical elements aligned vertically in triangular pitch. Coolant, at 23.8 atm., enters reactor at 660°F, cools reflector, flows upward in tricuspid spaces between fuel elements, exits at 1380°F. Control: 36 control rods of B_4C dispersed in graphite matrix; ZrB_2 and Rh, which enhances negative temperature coefficient, dispersed in graphite matrices for burnable poisons.

Power: 115 MW(t); 40 MW(e).

Code:	0311	12	31716	44	5941	726	81111	921	105
				45			84677		

No. 13 TARGET* 1000-MW(e) Power Reactor (Possible Modification)

General Atomic

References: GA-4706; At. Energy Rev., 1, No. 3, pp. 119-139, Sept. 1963.

Originators: Staff members.

Status: Preliminary design, 1964; possible alternative fuel to allow release of fission products.

Details: Thermal neutrons, steady state, converter. Fuel-moderator-fertile material: pyrocarbon-coated particles (1 μ or less in diameter) of carbides of U^{235} and Th dispersed in graphite. Additional moderator: BeO. Coolant: helium. Reflector: graphite. Fuel element: main features include graphite fuel matrix around BeO spine in cylindrical graphite element with an upper reflector and an internal fission-product trap. Element is 4.5 in. diameter, 20 ft long. Helium flows upward through channels formed by the approx 5500 fuel elements and approx 240 control rods that are oriented vertically in triangular arrangement to form cylindrical core. Active core height, 15.5 ft; diameter, 31.1 ft. Coolant inlet temp., 720°F; outlet, 1470°F. Reactor vessel: prestressed concrete lined with steel, 56 ft, 6 in. ID, 76 ft, 8 in. outside height. Control: negative temperature coefficient; 242 vertical cylindrical rods, 3 in. by 20 ft; rod is stainless-steel sheath holding boron carbide in graphite (30 w/o boron) containers. Power: 2270 MW(t), 1000 MW(e).

Code: 0311 12 31716 44 5941 726 81111 921 105
84677

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* Thermal Advanced Reactor, Gas-cooled, Exploiting Thorium

No. 14 OECD* High Temperature Gas Cooled Reactor Project (DRAGON)

European Nuclear Energy Agency

References: Proc. Second U.N. Int. Conf. 9, pp. 289-305; Proc. Third U.N. Int. Conf., 1, pp. 318-325; Nucl. Power, 5, No. 46, pp. 112-117, Feb. 1960.

Originators: L.R. Shepherd et al.

Status: Critical Aug. 23, 1964.

Details: Thermal neutrons, steady state, breeder. Fuel: 14 kg U^{235}. Coolant: helium. Moderator: graphite. Fertile material: Th. Reflector: graphite. Core divided into two zones: central-zone fuel of UC_2-ThC_2 particles coated with pyrolytic carbon and silicon carbide; surrounding-zone fuel UC_2-ZrC particles coated with pyrolytic carbon. Fuel particles mixed with resin-bonded graphite powder and pressed into compacts, which are inserted into graphite fuel tubes. Seven tubes form a fuel element. Core composed of 37 fuel elements in triangular lattice forming approximate cylinder within annular graphite reflector. Surrounded by steel pressure vessel. Around pressure vessel are six branches; each contains at the top heat exchanger and blower. Inlet and outlet ducts arranged concentrically, and flow is baffled, so that pressure vessel is exposed only to coolant gas at inlet temperature. Coolant passes upward through core, then to inner concentric duct of one of the heat-exchanger branches, then goes to blower and to annular space between neutron shield and pressure vessel. In reactor experiment, steam produced is bled off, not used for power in turbines. Inlet temperature of coolant in reactor: 350°C (662°F); outlet temperature: 750°C (1382°F). Shielding: steel pressure vessel; water; concrete building around steel vessel. Control: 24 B_4C rods arranged in circle immediately surrounding core, 1 for fine control. Power: 20 MW(t). Power density: 14 MW/m^3.

Code: 0312 12 31716 44 5941 726 81111 921 105

* Organization for Economic Cooperation and Development

No. 15 Experimental Beryllium Oxide Reactor (EBOR)

General Atomic

References: Proc. Third U.N. Int. Conf., 6, pp. 323-332; Nucleonics, 19, No. 3, p. 31, March 1961; GA-2603.

Originators: Staff members.

Status: Expected to begin operation in 1966; land-based prototype for Maritime Gas-cooled reactor.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: 70% enriched U as UO_2 combined with BeO in compacts. Coolant: helium. Reflector: BeO. Core structure: fuel-moderator pellets, 0.33 in. diam., 0.43 in. long, contained in Hastelloy-X fuel pins, 178 pellets to a pin. Fuel elements contain spine of BeO, instrumentation tube, fuel pins, inlet and outlet ports for gas coolant, and shroud. 52 reflector elements are Hastelloy-X-clad annular BeO rings. 36 core elements arranged cylindrically. Core, 23.43 in. square, 76 in. long, is surrounded by 7 in. BeO. Coolant enters and leaves through two concentric passages at side of reactor. Gas flow divides on entering reactor. One stream flows upward between pressure vessel and thermal shield, then downward between core container and thermal shield. Other stream flows downward between pressure vessel and thermal shield. Streams join below core and flow upward through core and into an upper plenum before leaving reactor. Coolant inlet temp., 750°F; outlet temp., 1300°F; pressure 1120 psia. Cylindrical pressure vessel of low Cr-Mo alloy steel. Control: 4 cruciform shim-safety and regulating rods of dysprosium oxide in alumina tiles contained in Hastelloy-X structure. Power density (avg.): 14.4 kW/liter. Power: 10 MW(t).

Code: 0313 15 31716 43 5931 711 81134 921 105

No. 16 Maritime Gas-cooled Reactor (MGCR)

General Atomic

References: Nucleonics, 19, No. 3, p. 31, March 1961; GA-2603.

Originators: Staff members, General Dynamics Corp.

Status: Land-based prototype (EBOR) being constructed, June 1965.

Details: To be based on experience with EBOR. Power: 53-77 MW(t).

Code: 0313 15 31716 43 5931 711 81134 921 105

No. 17 Ultra High Temperature Reactor Experiment (UHTREX) (Formerly TURRET)

LASL

References: LA-2198; LAMS-2469; LAMS-2623; *J. Franklin Inst., Monograph No. 7*, pp. 127-132, 1960; *Trans. Am. Nucl. Soc.*, 2, No. 2, pp. 148-149, 1959; *Atomics*, Nov.-Dec., 1964, p. 13.

Originators: R.P. Hammond *et al.*

Status: Completion scheduled for 1966.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: 93% enriched U^{235} as carbide in graphite. Coolant: helium. Reflector: graphite; ungraphitized carbon. Hollow cylinders of commercial graphite, 1 in. OD, ½ in. ID, 5½ in. long, are impregnated with $UO_2(NO_3)_2$, then dried and heated to convert nitrate to oxide, predominently UO_2. Subsequent heating in core completes conversion to carbide. Carbide is present as microspheres coated with pyrolytic carbon. Coating delays release of fission products. Core: vertical graphite cylinder 70 in. diam. Fuel-moderator elements positioned in horizontal, radial holes, which pierce core at 15° intervals in 13 equally spaced horizontal planes, making a total of 312 channels. Core can be reloaded while operating at full power. Each channel contains 4 elements end to end. New element introduced through gas lock and rammed into core. Inner-most element falls down a slot into central core plug and is removed through gas lock and conveyor. Core mechanically rotated and indexed at 15° intervals so each vertical row of fuel elements is aligned in turn with the single row of stationary loading rams. Original reactor name, TURRET, came from this feature. Coolant enters core at bottom, at 1600°F, through 12.5 in. diam. annulus; flow divides into multiple radial fuel channels, and is collected in plenum at core periphery for discharge from reactor at 2400°F. Core surrounded by reflector of 4 in. graphite and 12 in. dense, ungraphitized carbon. Containment: 14 ft diam. carbon-steel spherical pressure vessel. Temperature held to about 600°F by internal refractory insulation of porous carbon brick. Conducted heat radiated to water-cooled shielding walls. Control: large negative temperature coefficient of reactivity allows fuel loading and coolant flow to control temperature and power during reactor operation; Mo tubes filled with ZrB_2 inserted into stationary part of core to compensate for about 20% δ k/k. Power: 3 MW(t).

Code: 0313 12 31716 44 5942 711 84677 921 105
81211

No. 18 Australian High Temperature Gas-cooled Reactor (HTGC)

Australian Atomic Energy Commission

References: J. Franklin Inst., Monograph No. 7, pp. 81-86, 1960; Nucl. Eng., 9, No. 92, p. 9, Jan. 1964; Proc. Third U.N. Int. Conf., 11, pp. 329-340; J. Nucl. Materials, 14, pp. 29-40, 1964.

Originators: Staff members.

Status: Design studies in progress.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator-fertile material: pebbles in which PuO_2 and ThO_2 are dispersed in BeO. Coolant: CO_2. Reflector: side, graphite; end, BeO. Core: cylindrical vessel containing pebbles, which are slowly added at top and removed at bottom. Containment: spherical pressure vessel. Coolant flows up between core and pressure vessel to top of core, then downward through core; leaves at bottom to go to heat exchanger. Coolant enters core at about 350°C (662°F); leaves at 700-800°C (1292-1472°F). Fuel surface temperature: 1100°C (2012°F) max. Coolant gas pressure: > 500 psi. Control: 2 vertical rods. Power density: 10-20 kW/liter.

Code: 0312 15 31717 46 5932 726 8111X 921 105

No. 19 Solid Homogeneous Reactor for MHD (Magnetohydrodynamic) Power Generation

Westinghouse Electric Corporation

Reference: Nucl. Energy, March, 1963, pp. 60-66.

Originators: E.J. Sternglass, T.C. Tsu, G.L. Griffith, and J.H. Wright.

Status: Preliminary design, March 1963.

Details: Thermal neutrons, steady state, converter. Fuel-moderator: slightly enriched U as UC_2 particles distributed uniformly in matrix of graphite. Coolant: helium. Reflector: graphite. Core: cylinder of graphite moderator and reflector, with vertical channels for fuel-moderator elements. Coolant enters bottom of core at 1690°F, passes upward through it, and leaves at 2500°F to go to a MHD generator. Fueling machine permits refueling of reactor while on load. Power: 600 MW(t); 264 MW(e).

Code: 0311 12 31716 42 5942 723 8XXXX 921 105

No. 20 Boiling-water Reactor With Solid Homogeneous Core

Brown, Boveri & Cie., Aktiengesellschaft

Reference: United Kingdom Patent 854,291.

Originators: Staff members.

Status: Patent granted Nov. 16, 1960; applied for Aug. 4, 1956 in Germany.

Details: Thermal neutrons, steady state, breeder. Fuel-moderator-fertile material: rods containing all three; typical mixture would be $U^{233}O_2$, BeO, and ThO_2. Coolant: boiling H_2O, only in quantity to flow around fuel elements to remove heat by boiling but not enough to act as moderator. Reducing amount of water reduces absorption of thermal neutrons and gives greater breeding efficiency.

Code:	0312	15	32101	45	5432	726	8XXXX	9XXX	105
		1X		4X	5XX2	72X			

No. 21 Semi-homogeneous Bismuth Cooled Breeder Reactor (SHR)

Japan Atomic Energy Research Institute

Reference: IAEA Symp. on Power Reactor Experiments, I, pp. 149-54.

Originator: I. Inoue et al.

Status: Design, 1962.

Details: Same as Semi-homogeneous Gas Cooled Breeder Reactor described in Data Sheet No. 11, except that liquid Bi rather than helium is used as coolant.

Code:	0312	12	31105	43	5131	726	8XXXX	941	105
				45	5941				

No. 22 Terrestial Unattended Reactor Power System (TURPS)

Martin Nuclear Div., Martin-Marietta Corp.

References: Nucleonics, 23, No. 8, p. 46, Aug. 1965; Trans. Am. Nucl. Soc., 6, pp. 320-321, Nov. 1963.

Originators: C.E. Murphy, staff members.

Status: Under investigation, 1965; USAF contract.

Details: Thermal neutrons steady state, probably burner. Fuel-moderator: U (probably enriched) mixed with ZrH_x. Originally, enriched U as a fuel and H_2O as moderator were suggested as alternatives. Coolant: boiling P_4S_3. Fuel-moderator of U-ZrH_x and segment of unfueled ZrH_x make up 2-segment element. Hydrogen migrates from unfueled segment to fueled segment to act as moderator. Control: reduction in heat load on unfueled segment reduces hydrogen migration, with consequent reduction in moderation and activity. Coolant in natural circulation boiling-condensing cycle. Temperature: 1000°F. Power: 100 kW(e) minimum.

Code: 0313 17 32113 44 5981 711 84677 9XX 105

No. 23 Solid Homogeneous Reactor for Rocket or Aircraft Propulsion

("Cabbage Head")

References: U.S. Patents 2,894,891 and 2,917,443; personal communication.

Originator: J.J. Grebe.

Status: Patents granted July 14, 1959, Dec. 15, 1959; first patent filed Oct. 3, 1949.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: U^{233}, U^{235}, or Pu^{239} dispersed in LiH, BeO, BeH_2, Be_2C, or graphite. Coolant: H_2 or air. No reflector. Fuel-moderator: tapered segments of two layers separated by coolant passages. At one end of the core, each segment layer converges and closes. At other end, each segment is joined to the adjacent segment. Fuel-moderator is porous to gas flow. Segments are bowed outwardly to form a roughly spherical or pear-shaped core, the outer segments being larger than the inner ones. Outlet ducts from the elements merge into a central coolant channel, which itself becomes larger and leads into a jet nozzle. The coolant flows into inlet ducts, passes through the porous fuel-moderator into coolant channels, and leaves through the central coolant channel and the nozzle. Coolant could enter as a liquid, vaporize as it is heated in passing through core, then dissociate to atomic state in further heating. Outlet temp.: over 6000°F max. for hydrogen. Porous structure designed to: permit close contact of coolant gas with fissionable material; and provide many short parallel paths for gas flow through it, to allow large volume of gas to pass through in unit of time without excessive pressure loss. The reactor could be divided into separate sections, according to the temperature gradient in each section, with different materials used in each section. At the hot (exit) end, a refractory material, such as graphite, would be used. For the cooler sections (the inlet and the central portions) less refractory moderators with better nuclear characteristics would be used. Li^7H or Be_2C would be examples. Control: B or Cd control rod may be used.

Code:	0313	12	31714	44	5982	711	81X11	9XX	105
		15	31715	45			81X12		
		17	32115	46					
			33115						

No. 24 Solid Homogeneous Reactor Moderated with Zirconium Hydride

Bendix Aviation Corporation

Reference: Unpublished reports, Bendix Corp., May 15, 1952.

Originators: Staff members.

Status: Preliminary design.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: U^{235} distributed homogeneously in ZrH_2. Coolant: H_2O. Reflector: H_2O. Fuel-moderator compacts distributed in insulated spherical core, which also contains tubes for coolant and for superheat. H_2O (under 150 psi) reflector, 2 in. thick, around core insulation; core 12.6 in. diam. Reactor shell 17.3 in. OD. Coolant enters at 280°F, 625 psi, leaves at 600°F, 600 psi. Control: Cd absorbing material in reflector. Power: 12 kW.

Code: 0313 17 31101 44 5982 711 81X62 921 105

No. 25 XMA-I Reactor

General Electric Co., Aircraft Nuclear Propulsion Dept.

Reference: G.C. Leeth, unpublished report, General Electric Co., 1958.

Originator: G.C. Leeth.

Status: Program cancelled, 1961.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: UO_2 (highly enriched) in BeO; yttrium hydride. Coolant: air. Core arrangement: UO_2-BeO and yttrium hydride arranged homogeneously in concentric polygons. Coolant leaves at 1800-2000°F. Moderator wall temperature: 1900°F; fuel-element temperature: 2500°F. Control: flat plate inserted radially between yttrium hydride slabs.

Code: 0313 110 31714 44 5932 711 8132X 9XX 105

No. 26 Pratt & Whitney Aircraft Reactor-II (PWAR-II)

Pratt & Whitney Aircraft, Div. of United Aircraft Corp.

Reference: Unpublished report, Pratt & Whitney Aircraft, 1960.

Originators: J. Levengood, J. Dissler, and J. Kalinowski.

Status: Program cancelled, 1961.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: 20% enriched U in UO_2 contained in BeO matrix. Coolant: Li. Reflector: BeO. Fuel-moderator in form of pins in hexagonal containers. Core: right circular cylinder contained in Nb-Zr pressure vessel. BeO reflector on sides and ends of core. Control: B_4C contained in 8 rotating drums. Power: 510 MW(t); similar reactor designed to produce 200 MW(t).

Code: 0313 15 31106 43 5931 711 81441 921 105

No. 27 Hydrogen-Cooled, Solid Homogeneous Reactor for Rocket Propulsion

Thiokol Chemical Corp.

Reference: Nucl. Sci. Eng., 7, pp. 377-386, April 1960.

Originators: J.J. Newgard and M.M. Leval.

Status: General design concept, 1960.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: uranium-loaded graphite plates. Coolant: hydrogen. Reflector: BeO. Core formed by parallel plates of uranium-loaded graphite, which may be clad or unclad. Plates spaced by protrusions and are tied together in bundles, 2 in. by 2 in. by 6 in., by W or Mo wire. Reflector: 6 in. thick, BeO blocks across top of reactor and along length of core. Axial holes for coolant flow through reflector. Coolant flows through top end section of reflector, through core (where temperature is raised to 4200°F), and is exhausted to nozzle. Small amounts of gas pass through reflector and through structural columns in core. Control: 18 curved control plates in reflector contain inside section of borated stainless steel, enclosed in stainless steel; space between them for coolant gas. Plates supported by tube. Rotating tube changes degree of neutron absorption.

Code: 0313 12 31715 44 59X2 711 81441 921 105

No. 28 Tory IIA-1 Reactor

Lawrence Radiation Laboratory, Livermore

References: UCRL-6923; Nuclear News, 7, No. 7, pp. 8-11, July 1964.

Originators: Staff members.

Status: Part of PLUTO program on nuclear ramjet engines; successful test of reactor, May 1961; program suspended, 1964.

Details: Intermediate neutrons, steady state, burner. Fuel-moderator: highly enriched UO_2 homogeneously mixed with BeO to form hexagonal elements. Coolant: air; exit-gas temperature, 1975°F. Reflector: graphite. Core structure: 100,000 elements (4 in. long, 0.297 in. across flats, with 0.200 in. diameter hole for passage of air) arranged in hexagonal bundles, about 5 in. across, within BeO structural elements to make cylindrical core of 32 in. diameter and 4 ft long. Air-cooled support tube in each corner of bundles. Around core are shroud, Al pressure shell, and 2-ft-thick graphite reflector. Reflector contains 8 graphite cylinders with boron steel at outer edge of one quadrant of each. Reflector is water-cooled. Air coolant flows in through inlet diffuser, into front of core, out core to exhaust. Control: cylinders rotate to control reactor by increasing or decreasing effective size of reflector; 4 rods, for fast control, near inner wall of reflector. Power: 155 MW(t).

Code: 0213 15 31714 44 5932 711 82441 921 105
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No. 29 Tory IIC Reactor

Lawrence Radiation Laboratory, Livermore

Reference: Nuclear News, 7, No. 7, pp. 8-11, July 1964.

Originators: Staff members.

Status: Part of PLUTO program on nuclear ramjet engines; successful test of reactor, May 1964; program suspended 1964.

Details: Very similar to Tory IIA-1 reactor. Several hundred thousand fuel-moderator elements make up core. Control rods within core. Operating temperature: > 2000°F. Power: 600 MW(t).

Code: 0213 15 31714 44 5932 711 8211 921 105

No. 30 SNAP Experimental Reactor (SER)

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

References: Nucleonics, 19, No. 4, pp. 73-76, April 1961; Proc. Third U.N. Conf. on Peaceful Uses of Atomic Energy, 15, pp. 164-175.

Originators: Staff members.

Status: Successful operation, 1959-1960; dismantled.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: fully enriched U^{235} in homogeneous mixture with ZrH_x (7 w/o U^{235}); 2.9 kg U^{235}. Coolant: NaK; inlet temp., 1000°F; outlet, 1200°F. Reflector: axial, 1.5 in. Be; radial, 3 in. Be. Core structure: 61 stainless-steel-clad fuel-moderator elements in triangular pattern to make up hexagonal core 9 in. across corners and 8 in. across flats. Rods 14 in. long and 1 in. in diameter. Core volume: ~0.4 cu. ft. Core vessel: 16 in. high, 9.5 in. diameter. Control: 3 Be drums rotate around core; reflector thickness varied; 3 Be safety plates fall away from reflector for scram. Power: 50 kW(t).

Code: 0313 17 31204 44 5981 711 82448 921 105

No. 31 SNAP-2 Development Reactor (S2DR)

Atomics International

References: Proc. Third U.N. Conf. on Peaceful Uses of Atomic Energy, 15, pp. 164-175; Nucleonics, 23, No. 6, pp. 44-47, June 1965.

Originators: Staff members.

Status: Test operation; predecessor to cancelled SNAP-2 Reactor.

Details: Very similar to SER, but with differences in dimensions, etc. 10 w/o U^{235} in fuel-moderator element; 37 fuel-moderator elements. Cladding: Hastelloy-N. Core loading: 4.3 kg U^{235}. Core vol.: ~0.4 cu. ft. Control: 2 Be drums; 2 Be safety plates. Power: 50 kW(t).

Code: 0313 17 31204 44 5981 711 82448 921 105

No. 32 SNAP-2 Reactor

Atomics International

Reference: Nucleonics, 19, No. 4, pp. 73-76, April 1961.

Originators: Staff members.

Status: Design intended as predecessor for SNAP-8; discontinued, 1963.

Details: Same as SNAP-8 (see below), but with power of 3 kW(e). Mercury Rankine conversion cycle.

Code: 0313 17 31204 44 5911 711 82448 921 105

No. 33 SNAP-8 Reactor

Atomics International

References: Proc. Third U.N. Conf. on Peaceful Uses of Atomic Energy, 15, pp. 164-175; Nucleonics, 19, No. 4, pp. 73-76, April 1961; 23, No. 6, pp. 44-47, June 1965.

Originators: Staff members.

Status: Test operation of nuclear prototype continuing in 1965.

Details: Very similar to SER. 10 w/o U^{235} in fuel-moderator element. 6.56 kg U^{235}. 211 fuel-moderator elements. Cladding: Hastelloy-N. Coolant inlet temperature: 1100°F; outlet, 1300°F. No axial reflector; 3 in. Be for radial. 6 control drums. Power: 600 kW(t); 35-70 kW(e), with mercury Rankine cycle.

Code: 0313 17 31204 44 5911 711 82448 921 105

No. 34 SNAP-10 Reactor

Atomics International

References: NAA-SR-3473; personal communication, R.A. Du Val.
Originators: Staff members.
Status: Preliminary design, 1959; terminated 1960.
Details: Thermal neutrons, steady state, burner. Fuel-moderator: U-ZrH_x, containing 3.2 kg U^{235}, in flat plates, 7 in. diam., 3/4 in. thick, stacked to form cylindrical core. Be plates, 1/8 in. thick, alternating with fuel-moderator plates in stacks to promote radial heat transfer. Cooling by direct conduction of heat from core to thermoelectric converter devices on face of reactor; cold-junction temperature: 780°F; hot-junction temperature: 1100°F. Reflector: 2½ in. Be on sides and ends. Reactor 12 in. diam., 18 in. high. Control: strong negative temperature coefficient; for startup or shut-down, two halves of reactor brought together or separated. Power: 12 kW(t); 0.3 kW(e).

Code:	0313	17	2122	44	5981	711	83169	921	109
							84677		

No. 35 SNAP-10A Reactor

Atomics International

References: Proc. Third U.N. Conf. on Peaceful Uses of Atomic Energy, 15, pp. 164-175; Nucleonics, 23, No. 6, pp. 44-47, June 1965; personal communication, R.A. Du Val.
Originators: Staff members.
Status: One reactor orbited in spacecraft April 3, 1965; another operated in ground test for 10,000 hours from Jan. 1965 to March 1966.
Details: Based on SNAP-2 and SNAP-8. Thermal neutrons, steady state, burner. Fuel-moderator: 10 w/o fully enriched U^{235} in homogeneous mixture with ZrH_x. Coolant: NaK. Reflector: Be. 37 fuel elements (1¼ in. diam.) in triangular array form hexagonal core, with Be plates in spaces between core angles and cylindrical core vessel (9 in. diam.). Coolant flows axially in spaces between fuel elements. Inlet temp., 900°F; outlet, 1010°F. External reflector of Be. Control: rotating four sections of reflector varies neutron leakage for startup control; burnable poisons; negative temperature coefficient for long-term control. Power: 34 kW(t), 0.5 kW(e). Thermoelectric direct power conversion system.

Code:	0313	17	31204	44	5981	711	82448	921	105
							84677		

No. 36 Hydride Moderated Boiling Reactor (SNAP-4).

Atomics International

References: Unpublished report, Feb. 18, 1963; SNAP Fact Sheet, USAEC, June 15, 1962.

Originators: Staff members.

Status: Development incorporated into later work.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: highly enriched U alloyed with hydrided zirconium in form of rods clad with noncorrosive alloy. Coolant: boiling H_2O at 1200 psi. Bundles of fuel-moderator rods make up vertical hexagonal core. Water enters bottom of reactor, is heated to boiling, and passes out of top to go through steam separator to turbine in direct-cycle operation. Control: rotating drums in reflector around core. Power: 1-4 MW(e).

Code: 0313 17 32101 44 5981 711 82448 921 105

No. 37 KIWI-A Reactor

LASL

References: Nucleonics, 19, No. 4, pp. 77-79, April 1961; Astronautics and Aeronautics, 3, No. 6, pp. 42-46, June 1965; LADC-5261.

Originators: Staff members.

Status: Tested, 1959-60; incorporated in later designs.

Details: Thermal neutrons, steady state, burner. Fuel-moderator: graphite plates in which $U^{235}O_2$ is incorporated. Coolant: H_2. Reflector: graphite. Core arrangement: fuel plates, 8 in. long, 5-8 in. wide, and ¼ in. thick, with longitudinal ribs on one surface. Plates stacked in cylindrical boxes, with ribs maintaining passage for flow of gas. Graphite disc, with coolant passages acts as neutron reflector at core inlet side. Fuel elements in annular zone around central island containing D_2O and control rods. Surrounding fuel zone is graphite flow-separator cylinder, graphite outer reflector, and Al double-walled pressure vessel, which is water-cooled. Pressurized H_2 goes into reactor via a plenum at upper end of reflector, passes through holes in reflector down into core inlet plenum, passes up between fuel plates, enters core-exhaust plenum, and leaves through the nozzle. Control: vertical scram and shim rods in central island. Exit-gas temperature: 3200°R. Power: 70 MW(t).

Code: 0313 12 31715 44 5931 711 8111X 923 105

No. 38 KIWI-B Reactor

LASL

References: Astronautics and Aeronautics, 3, No. 6, pp. 42-46, June 1965; LADC-5490; IEE Trans. in Nucl. Sci., NS-12, No. 1, pp. 160-168, Feb. 1965.

Originators: Staff members.

Status: Several designs tested.

Details: Similar to Kiwi-A. Fuel: minute beads of uranium carbide coated with pyrolytic carbon. Coolant: liquid H_2. Control: drums in reflector.

Code: 0313 12 31115 44 5931 711 81441 923 105

No. 39 NRX-A Reactor

Westinghouse Astronuclear Laboratory, Westinghouse Electric Co.

Reference: Unpublished report, Nov. 1963.

Originators: Staff members; design developed from LASL Kiwi-B concepts.

Status: Engineering design, 1963; development continuing.

Details: Thermal neutrons, steady state, burner. Fuel: enriched U. Moderator: graphite. Coolant: H_2. Reflector: inner, graphite; outer, Be. Fuel elements: enriched UC_2 coated with pyrolytic carbon and uniformly dispersed in graphite. NbC coating on coolant passages and on parts of external surface. Fuel elements grouped to form clusters. Pressure vessel around outer reflector. Shield between pressure vessel and propellant tank. Control: 12 cylindrical Be control drums within outer reflector rotate poison strip (20% B^{10} in Al) toward or away from core; shim rods in outer reflector; shim rods in some unfueled elements in core--moderator shims of graphite and poison shims of graphite containing dispersed Ta carbide.

Code: 0313 12 31715 44 5942 711 81441 921 105
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