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GAS-COOLED REACTORS

by

William H Richardson

and

Frances Strachwitz

September 1959

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GAS-COOLED REACTORS

Introduction

This compilation contains 517 references to the unclassified literature on gas-cooled reactors to May 1959. It includes articles, reports, books, and patents. Reactors cooled by air are included, as well as all types of reactors by function--power reactors, research reactors, design studies, and prototypes.

The references are arranged by subject in the same order as in the Contents. The references are indexed by reactor, author, report number, and Geneva Conference Paper number.

The Gas-Cooled Reactor Directory lists all named reactor projects by type. It gives the status, location, contractor, and basic characteristics of each project.

1. CLOSED-CYCLE POWER

1 - 1

LE CHOIX DU CYCLE THERMIQUE DANS UNE USINE GENERATRICE NUCLEAIRE
Chambadal, P.
Revue Generale de l'Electricite, Vol. 67 No. 6, p. 332-45. June 1958.

Choice of heat cycle in nuclear power plant; analysis of heat cycles in existing reactors using water vapor as motor force, with or without heat exchanger; as such use of water is limited by moderated value of heat source, author reopens discussion on using gas, such as helium, nitrogen or carbon anhydride, for reactor cooling which, after having absorbed heat from reactor, is used in turbine.

1 - 2

CLOSED-CYCLE-GAS-TURBINE-REACTOR-POWER-PLANT
Winkler, W.
(NP-6652), Reaktor AG, Wurenlingen, Switzerland
November 25, 1957. 15 p.

A discussion of different problems involved with the development program of a closed-cycle-gas-turbine-reactor-power-plant is given. The types of enriched uranium fuel elements which permit an operation at temperatures of about 700°C, and some of the development work done in this field by private industry in the United States are discussed. Investigations on the choice of the gas from the nuclear and thermodynamic viewpoint are presented, and heat transfer and pressure loss problems for different gases and their mixtures are included. Calculations of gas activity due to neutron capture by the gas, impurities, residual matter from erosion or incomplete cleaning, and fission product leakage from the fuel elements are included. The thermal efficiency of different steam and gas cycles are also compared. (auth)

1 - 3

CONTROL ROD DRIVE DEVELOPMENT FOR CLOSED CYCLE GAS COOLED REACTOR PROGRAM

(FICO-106), Ford Instrument Co., Long Island City, N.Y.
March 1958. 30p., Contract AT(30-3)-250.

The development and test program undertaken to produce an economical control rod drive mechanism suited to the requirements of a high temperature gas cooled reactor is described. The system design features are described as well as the behavior of the specially selected materials during the testing in a high temperature non-radioactive environment.
(auth)

1 - 4

15 MW GAS COOLED CLOSED-CYCLE REACTOR POWER SYSTEM STUDY

(FICO-101) Vols. 1 and 2. Ford Instrument Co., New York
Final Report. April 1, 1957. Contract AT(30-3)-215.

This is to be a thermal, heterogeneous, nitrogen-cooled, graphite-moderated, and 10% enriched UO_2 fueled reactor. The core is a right cylinder with end and side reflectors. The heat input for the power plant is 44 Mw, and the electrical output is 15 Mw. Theoretical studies covering the physics of the processes occurring in the core, the thermodynamics of heat transfer from the fuel elements to the coolant gas, and the thermodynamics of the gas cycle in the system as a whole were studied. Mechanical and engineering design studies were also made of the core materials, structural members, gas cooling equipment, and all other power plant accessories that must meet requirements unique to nuclear reactors. (M.H.R.)

1 - 5

GAS COOLANT FOR NUCLEAR REACTORS

Silverberg, M.

In Engineers Joint Council, Second Nuclear Engineering and Science Conference, Proceedings. New York, Pergamon Press, 1957. Vol. 2, p. 470-472.

The status of closed cycle gas-cooled reactor power plants is discussed. The advantages and disadvantages of this type of system are considered. Particular emphasis is placed upon some of the problem areas brought to light by design studies and their potential effect on the future of the gas-cooled reactor. (W.H.R.)

1 - 6

GAS TURBINE IN ATOMIC ENERGY

Perkins, J. E. B.

Nuclear Power, Vol. 1, No. 3, p. 108-13, July 1956.

Possible reactor turbine systems discussed and tentative evaluations made of various combinations of thermal and fast reactors with open and closed cycle turbines; of possible gases considered, air is considered difficult to excel; how gas turbine might be used for land, sea, and air application.

1 - 7

INFLUENCE OF WORKING-FLUID CHARACTERISTICS ON DESIGN OF CLOSED-CYCLE GAS TURBINE

Robinson, S. T.

Am. Soc. Mech. Engrs., Paper No. 57-GIP-13 for meeting March 18-21, 1957. 12 p.

Procurement of two closed cycle gas turbine plants is planned in United States, both suitable for use with gas cycle nuclear reactor as heat source; these plants differ widely in output, purpose and nature of working fluid; effect of nature and characteristics of working fluid on design of non-nuclear components.

1 - 8

NUCLEAR GAS ENGINE

Fraas, A. P.

(CF-58-9-12), Oak Ridge National Lab., Tenn.

September 25, 1958. 115p., Contract (W-7405-eng-26)

A preliminary design study of the nuclear gas engine, consisting of a gas-cooled reactor directly coupled to a reciprocating engine, is presented. The principles of operation of the proposed gas engine are outlined and typical variations are discussed. The nuclear gas engine is compared with other reciprocating engines and air compressors. A comparison between the ideal and actual cycles is made, with particular attention given to applications and development of the nuclear gas engine are discussed. (W.D.M.)

1 - 9

NUCLEAR POWERED GAS TURBINES FOR LIGHT WEIGHT POWER PLANTS

Hammit, F. G. and Ohlgren, H. A.

In Engineers Joint Council, Second Nuclear Engineering and Science Conference, Proceedings. New York, Pergamon Press, 1957. Vol. 2, p. 2-25.

The suitability of closed-cycle gas turbine power plants in combination with various types of nuclear reactors is examined. Typical examples of both a heterogeneous and a homogeneous reactor power plant are presented. Weight, cost, and performance are compared for various possible fluids over a range of temperature and pressure. Comparisons are made with alternative heat engine systems. (W.H.R.)

1 - 10

PROGRESS ON MATERIALS

- - -

Key to Gas-Cycle Reactor Potential. Nucleonics Vol. 14, No. 3, p. 42-4, March 1956.

While Oak Ridge and Brookhaven gas-cooled research reactors are not suitable for power production, pressurization of coolant gives such reactors new attractiveness for power purposes; materials problem posed by turbine gases at 1200-1500 F; possible materials to serve as working gas, fuel elements and moderator; comparison of suggested single loop open and closed gas cycle reactors; problems remaining to be solved.

1 - 11

SKETCH OF A GAS TURBINE POWER PLANT

Young, G.

(CF-3290) Chicago Univ. Metallurgical Lab.

August 10, 1945. Decl. March 12, 1957. 14p. (Contract W-7401-eng-37.)

Some calculations are made for a Be metal pile cooled by He operating in a gas turbine cycle to give mechanical power output. An efficiency of about 20% is indicated. (auth)

SMALL GAS-CYCLE REACTOR OFFERS ECONOMIC PROMISE

Daniels, F.

Nucleonics, Vol. 14, No. 3, p. 34-41, March 1956.

The design of a 20MW helium-cooled graphite moderated reactor operating a 5MW gas turbine is considered in detail. The core, 6 ft in diameter and 5 ft 5 in. high, is an assembly of perforated graphite blocks. It contains 13.7 kg. U^{235} in the form of UC_2 (10% U^{235}) powder or pellets in graphite cartridges. Control rods are B or B_4C in Mo cans. Helium at 225 lb/in.² and 1350°F, from the reactor, operates the turbine via a heat exchanger. The preparation and processing of the fuel are considered. The reactor has good neutron economy and should be inherently safe.

2. CONSTRUCTION

2 - 1 BERKELEY—PROGRESS REPORT

- - -

Nuclear Power. Vol. 2, No. 12, April 1957. p. 142-3, 1 supp. chart; see also Engineer Vol. 203, No. 5270, 5292 January 25, 1957. p. 143-5, June 28, p. 990-1; Civ. and Structural Engrs. Rev. Vol. 11, No. 3, March 1957. p. 127-9.

Status of project at site in Gloucestershire where one of Central Electricity Authority's first nuclear power stations is under construction by AEI-John Thompson Nuclear Energy Co. Ltd; station will comprise two natural uranium, graphite moderated gas cooled reactors supplying four 85Mw turbines; total sent out capacity will not be less than 275Mw and will probably approach 300Mw; notes concerning plant excavations; perspective cutaway color diagram of station.

2 - 2 BRADWELL NUCLEAR POWER STATION

- - -

Civ. & Structural Engrs. Rev. Vol. 11, No. 4, p. 190-1, April 1957.

station will occupy portion of site of approximately 142 acres and when completed will have output of 300 Mw; plant will consist of two natural uranium, CO₂ gas cooled, graphite moderated reactors supplying heat to 12 boiler units; six boilers will be associated with each reactor.

2 - 3 HOW THEY ARE BUILDING NUCLEAR BRITAIN Calder, N. New Scientist 4, 413-14, 16, 18(1958).

The progress and differing methods in the building of the Bradwell, Berkeley, Hunterston, and Hinkley Point power reactors are discussed. (W.D.M.)

2 - 4

PROGRESS AT HUNTERSTON NUCLEAR POWER STATION

- - -

Engineer Vol. 206, No. 5346, July 11, 1958. p. 51-4.

Construction work on station at Hunterston on Ayrshire coast; excavation for foundations of second reactor; station will have installed capacity of 360 mw and ultimate net electrical output of 320 mw; particulars of 300-ton Goliath crane with 30-ton auxiliary crane, employed in erection.

2 - 5

WELDING AND NUCLEAR POWER PROGRAMME--ANNUAL LECTURE 1959

Owen, L.

Brit Welding J. Vol. 6, No. 5, p. 197-204, May 1959.

Welding problems with regard to fuel elements and pressure vessel of Calder Hall type of reactor; welding of beryllium used in advanced gas cooled reactor; welding involved in components for fast fission breeder reactor; it is concluded that civil reactors to date have not called for any basically new material but demand higher standards of existing materials and practices.

3. CONTROL SYSTEMS

3 - 1

AUTOMATIC CONTROL OF A NUCLEAR REACTOR--APPLICATION TO THE SACLAY PILE

Bonnaure, P., Braffort, P., Pelchowitch, I. and Weill, J.
J. Nuclear Energy. Vol. 1, p. 24-38, August 1954. (In French)

It is actuated by an error signal corrected by a signal which is proportional to the reactivity. It is demonstrated that such a system is stable. Experimental results obtained with the Saclay pile are presented and are shown to agree with the theory. Stability of better than 0.1% is easily achieved in steady runs, as well as safety when power is changed.

3 - 2

CALIBRATION OF CONTROL AND SHUTOFF RODS IN GLEEP BY RELAXATION TIME MEASUREMENTS

(AERE-N/R-130) Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.
December 5, 1947. Decl. April 18, 1958. 13p.

Measurements were made to determine the effectiveness of the control and shutoff rods in Gleep. The changes in the reproduction factor of the pile, brought about by these rods, were determined by relaxation time measurements. Some preliminary information was also obtained on the effects of pressure and temperature changes on the reproduction factor. (auth)

3 - 3

CONTROL OF THE G-2 AND G-3 REACTORS

Ertaud, A. and Pages, A.
Bull. inform. sci. et. tech. No. 20, p. 126-39, August 1958. (In French)

A brief description is given of the ensemble of mechanisms which permits control of the reactor, gives information to the personnel in charge of the reactor operation, and monitors the safety of the personnel and materials. (J.S.R.)

3 - 4

CONTROL OF THE SACLAY REACTOR

Weill, J.

L'Onde elect. 33:692-701, December 1953. (In French)

Describes the Saclay reactor which uses uranium rods plunged vertically into heavy water. Ionization chambers of the boron-carbide type are used for measurement of the reactor power by means of the slow neutrons produced. The ionization chambers are connected to pre-amplifiers using electrometer valves and thence via long cables to d.c. amplifiers with overall feedback. A circuit diagram of the arrangement is given and safety arrangements are described. Automatic equipment indicates the level of the heavy water, thus permitting accurate control of the amount required. The positions of the plates for regulating the activity of the pile are adjusted from the control console by means of a servo-mechanism system. Ionization chambers with associated amplifiers are also used for measurement of the pile reactivity and of the ionization at 7 points in the hall containing the pile. The general arrangements for temperature and loss measurements are shown in diagrams.

3 - 5

CONTROL IN SOME TYPES OF NUCLEAR POWER REACTORS

Tescari, A.

Energia nucleare (Milan) 5, 117-30, February 1958. (In Italian)

This paper deals with regulation and control systems in nuclear power reactors. Only two types of reactors are considered, owing to the limited time available: the natural uranium reactor, moderated by graphite and cooled by CO₂, and the enriched uranium reactor, moderated and cooled by pressurized natural water. After a brief revision of such reactors characteristics, the author examines the regulation and control problems from a general point of view although some data are also reported referring to the English 200 electric Mw power reactor and the American PWR ALCO 10 electric Mw power reactor. A brief hint is also made to electronic and nuclear equipment. (auth)

3 - 6

THE CONTROL SYSTEM OF THE WINDSCALE PILES

Smart, E. and Gallie, R. R.

(RDB(W)/TN-80), Gt. Brit. Windscale Works, Sellafield, Cumb., England
July 1953. 20p.

The information available on the control system of the Windscale reactors is discussed, and recommendations are made for experiments which are necessary to give fuller information. (auth)

3 - 7

EFFECTIVENESS OF NEW EMERGENCY S.O. RODS IN B.E.P.O.

Horton, C. C.

(BCM/WP-2), Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England. October 26, 1953. 2p.

The effect of thermal link and spring actuated emergency control rods in BEPO is calculated. For 4 rods of 3-in. diameter and 10-ft length placed in the corner holes of the thermal column, the control expected is 1.8% at least. For 12 rods of 1 3/8-in. O.D. and 13-ft length placed symmetrically on the circumference of a circle of 72.5-in. (10 pitches) radius horizontally, the control expected is 2.1% at most. (auth)

3 - 8

ELECTRO-MECHANICAL INSTALLATIONS

Lignieres, M., Bobigat, J. and Darre, A.

Bull. inform. sci. et tech. No. 20, pp. 140-51, August 1958.
(In French)

The principles at the basis of the studies on the operation and control and the electronic Ranical devices of the G-2 and G-3 reactors and the power station are reviewed. (J.S.R.)

3 - 9

ESTIMATES OF THE EFFECTIVENESS OF THE CONTROL AND SAFETY RODS FOR GLEEP

Rennie, C. A.

(AERE-T/R-114), Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England. July 1947. Decl. April 16, 1958. 8p.

Estimates are made of the change in the multiplication factor as controlled by the control and safety rods in GLEEP. The case of a rod perpendicular to the axis of the pile was considered as it is simpler to treat theoretically. Also the reacting core was assumed to be homogeneous. Curves showing the effectiveness of a rod of given dimensions as a function of its distance from the curve of the pile are given. (J.S.R.)

- 3 - 10 IMPROVEMENTS IN OR RELATING TO NUCLEAR REACTOR CONTROL MECHANISM.
- - -, To Rolls-Royce, Ltd.
British Patent 803,708. October 29, 1958. Nuclear Power 3,
622(1958) December.
- Improvements for a reactor control system have been proposed by Rolls-Royce. The system is space saving because the control elements do not protrude from the reactor when removed. Schematics are shown of the new system. (J.H.M.)
- 3 - 11 KINETIC AND CONTROL OF THE BERKELEY REACTORS
Yellowlees, J. M. and French, P. R. J. (AEI-John Thompson Nuclear Energy Co., Ltd., Knutsford, Ches., Eng.).
A/CONF.15/P/55
- Requirements are listed for the over-all plant load control system. To meet these conditions the design of the over-all control system must be coordinated with that of the plant mechanical details, reactor protection system, and station electrical system. The design of the control and protection system of the Berkeley Nuclear Power Station is summarized, and certain characteristic reactor transient conditions are described. (W.D.M.)
- 3 - 12 MANUAL OF INSTRUMENTS AND CONTROLS FOR THE BROOKHAVEN NUCLEAR REACTOR
(M-4416) Massachusetts Inst. of Tech., Cambridge.
Servomechanisms Lab. and Jackson and Moreland (Engineers), New York
Book 3, Vol. 11. May 1949. Decl. January 11, 1956.
- This book is one of a series written for Brookhaven National Lab., prepared under contract to H. K. Ferguson Co.
- Reactor activity instrument systems designed for the Brookhaven Reactor are described. The instruments indicate reactor conditions and serve to guide the operator, actuate the alarm and emergency-shutdown systems, and record reactor-activity data. Designations and functions of the various systems are described. Photographs and circuit diagrams of the various instruments and components are included. (C.H.)
- 3 - 13 THE OVERALL CONTROL OF NUCLEAR POWER STATIONS OF THE GAS-COOLED THERMAL REACTOR TYPE
Smith, R. J., (General Electric Co. Ltd., Erith, Kent, Eng.).
A/CONF.15/P/85, 32p.

The location of control facilities is briefly discussed, and basic operational requirements are reviewed. The influence of the plant on part-load operating conditions and permissible transient performance is emphasized, and typical part-load characteristics are quoted. In a section on the main control elements, the choice of coolant circulator systems is discussed in some detail. Requirements to ensure trip-free operation and safe control apparatus are listed, and power cut-back advocated. Typical automatic control arrangements are analyzed. (auth)

- 3 - 14 LA REGULATION DE LA CENTRALE NUCLEAIRE E D F. 1. (Control of the EDF.1 Power Plant).
Deloux and Laurent (Electricite de France, Paris).
A/CONF.15/P/1200. 21p.

The EDF.1 nuclear power plant was designed to supply the French general network with electrical energy. It operates, as a rule, at full power, but it was designed to be as versatile as a classical steam plant. This consideration imposes the use of an automatic control system to assure its correct functioning in normal operation, the utilization of automatic sequences to reduce the charge, and the placing in service of auxiliaries in the case of an incident. A survey is given of the way in which the problem is presented, the means used to study the problem, and the results obtained with respect to the control system adopted and the operations to provide for conditions created by incidents. (J.S.R.)

- 3 - 15 SERVICE MACHINE AND CONTROL ROD DRIVES
Fieser, G. W., Nuclear Products-Erco Div., ACF Industries, Inc., Washington, D. C.
In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, TID-7564, October 21-22, 1958. p. 98-109.

A description is given of the service machine, including its approximate size and operating functions. The control rod drives and their components are discussed. The operations of the entire drive mechanisms are briefly described. (auth)

3 - 16

THE TRANSIENT POWER CHARACTERISTICS OF B.E.P.O. UNDER AN AUTOMATIC
POWER REGULATOR

Bowen, J. H.

(AERE-E/R-510) Gt. Brit. Atomic Energy Research Establishment, Harwell,
Berks, England.

April 1950. Decl. December 20, 1956. 21p.

A linear differential equation is experimentally obtained which approximately describes the pile and its associated control apparatus. A possible application is suggested and examples given. (auth)

4. COOLANTS

- 4 - 1 ACTIVITY IN THE COOLING AGENTS OF REACTORS AND THE DETECTION OF
ESCAPE OF FISSION PRODUCTS II
Labeyrie, J.
Bull. inform. sci. et tech. No. 14, pp. 2-13, February 1958.
(In French)

The normal activities to be found in the reactor coolants, such as air, light and heavy water, compressed CO_2 , organic fluids, and liquid sodium, are discussed. An examination of the procedures for the separation of fission product activity from the normal activity of the coolant is made. Coolants of air, light and heavy water, and compressed CO_2 are considered.

- 4 - 2 CHEMICAL PROBLEMS IN COOLANT GASES
Dennis, W. E. (G.E.C./Simon-Carves Atomic Energy Group).
Nuclear Eng. 2
p. 321-3, August 1957.

Processes involving the coolant gas such as the mass transfer of carbon, the oxidation of metals and the removal of moisture should not be viewed as isolated and independent phenomena. In the absence of further information on the effects of irradiation on chemical reactions occurring in the gas circuit only a qualitative picture can be obtained of the gas composition likely to be established during operation. Under what appear to be likely conditions in a CO_2 cooled graphite moderated reactor it may be shown that significant concentrations of hydrogen might be produced by the reduction of H_2O with CO . The water-gas reaction will prevent the accumulation of hydrogen from extraneous sources but may render the removal of moisture more difficult. In the presence of very small concentrations of water vapor the formation of zirconium hydride is thermo-dynamically possible at all temperatures and cannot, therefore, be neglected in specifying conditions for oxidation tests. (auth)

- 4 - 3 COMPARISON OF GASES FOR USE AS THE COOLANT IN A GAS-COOLED REACTOR
Samuels, G.
(CR-58-4-108) Oak Ridge National Lab., Tenn.
April 28, 1958. 12p. Contract (W-7405-eng-26).

The expression $[M(\text{Pr})^{2/3} / (M \text{ Cp})^3]$ which is often used to compare the merits of various gas coolants is derived and the errors involved in using this term are discussed. An example using CO_2 and He is shown to illustrate the difference between the true pumping power required and that predicted by the above expression. (auth)

4 - 4

COOLANT CHOICE FOR V.H.T. REACTOR

Portescue, P.

Nuclear Power. Vol. 2, No. 17, p. 381-5, September 1957.

Heat transfer and thermodynamic considerations relating to gas cooled reactors for high temperature operations, possibly in conjunction with forms of gas turbine plant; particular reference made to solid homogeneous cored reactors; properties of working gas and pertinent mathematical relationships involving pumping power, etc, required in design of components; numerical results tabulated for number of different coolants.

4 - 5

EVALUATION OF COOLANTS AND MODERATORS FOR THE MARITIME GAS-COOLED REACTOR

- - -

(GA-570) General Dynamics Corp., Groton, Conn.

December 10, 1958. 463p. Contract AT(04-3)-187

The following systems were investigated, all using heterogeneous UO_2 with stainless-steel cladding: graphite-helium, graphite- CO_2 , light water- CO_2 , and zirconium hydride-carbon dioxide. The helium-cooled graphite-moderated system was selected as most promising for the MGCR.

4 - 6

GAS COOLED POWER REACTOR COOLANT CHOICE

Heacock, H. W. and Nightingale, R. E.

(HW-56362(Rev.)). General Electric Co., Hanford Atomic Products Operation, Richland, Wash.

June 12, 1958. 9p. Contract W-31-109-Eng-52.

The current status of helium and carbon dioxide technology is described in the light of the Gas Cooled Reactor Program requirements. The problem of containing high-pressure helium at high temperature is discussed, and it is concluded that, by proper attention to the design, construction and maintenance of a plant, a high degree of helium leak-tightness can be achieved at small additional cost when compared with a carbon dioxide system. What is more, the cost of making up helium losses in a practically achievable system is estimated to be small compared with other fixed and operating costs.

Graphite-carbon dioxide reaction data are reviewed. It is shown that carbon dioxide at atmospheric pressure and low flow rates should be compatible with a graphite moderator up to 525°C. No data are available at the high pressures and flow rates that would be encountered in power reactors. Significantly higher oxidation rates may result, however, perhaps limiting bulk moderator temperatures to 450 to 500°C. Improved carbon materials, protective coatings and inhibitors, and/or operating practices may be developed that will allow significant future increases in these limiting temperatures. (auth)

- 4 - 7 GASEOUS COOLANT FOR NUCLEAR REACTORS BASED ON THE INCREASED HEAT CAPACITY OF DISSOCIATED GASES
- - -, (to Siemens-Schuckertwerke AG., Germany).
German Patent DAS 1033 344. Atompraxis 4, 477(1958). (In German)

- 4 - 8 HEAT TRANSFER PROBLEM IN REACTOR TECHNOLOGY
Junkermann, W.
Nukleonik 1, pp. 73-87, July 1958. (In German)

Properties and characteristics of different coolants are described. Included are air, CO₂, He, Ne, N, and H.

- 4 - 9 PHYSICAL CONSTANTS OF PROPOSED COOLANTS
Bentley, R., Brown, G. and Schlegel, R.
(CP-3061) Chicago. Univ. Metallurgical Lab.
June 23, 1945. Decl. December 14, 1955. 48p. Contract W-7401-eng-37.

Physical constants, including melting and boiling points, density, heat capacity, thermal and electrical conductivities, viscosity, and vapor pressure are given for a number of proposed liquid and gaseous coolants. Summary tables for quantities which determine heat transfer characteristics are also given. (auth)

- 4 - 10 SELECTED VALUES OF THE PHYSICAL PROPERTIES OF VARIOUS MATERIALS
Tebo, F. J.
(ANL-5914) Argonne National Lab., Lemont, Ill.
September 1958. 261p. Contract W-31-109-eng-38.

Selected physical properties of various solids, liquids, and gases of interest in nuclear reactor heat transfer and fluid flow analysis are presented. In most cases, values for

thermal conductivity, specific heat, density, viscosity, and Prandtl numbers are given as functions of temperature. A table of room temperatures values of some properties was compiled on materials for which data as a function of temperature were not found. A short list of references to additional physical property data on many materials is provided at the end of the compilation. The values presented were compiled after a brief survey of the classified and unclassified literature. Much of the data came from sources most frequently used for reference such as Reactor Handbook and the Liquid Metals Handbook. (auth)

- 4 - 11 SELECTING THE REACTOR COOLANT. PART I. GENERAL CONSIDERATIONS.
Ablitt, J. F. (Atomic Energy Research Establishment, Harwell, Berks, Eng.).
Atomics and Nuclear Energy. 9:316-17, 323, September 1958.

The main points which must be considered in the search for a suitable coolant for a chosen fuel and moderator are examined. The cost factors involved are discussed.

- 4 - 12 SELECTING THE REACTOR COOLANT. PART II. SPECIFIC MATERIALS
Ablitt, J. F. (Atomic Energy Research Establishment, Harwell, Berks, Eng.).

Liquid metals, heavy and light water, gases, and organic compounds are considered for use as reactor coolants. The physical and chemical properties of these materials are examined, and applications are proposed by screening the various materials for desired qualities.

5. COOLING SYSTEMS

- 5 - 1 AN ANALYSIS OF THE CONTROL ROD COOLING PROBLEM WITH APPLICATION TO SAR GEOMETRIES
Millard, J. W.
(KAPL-M-S3G-RES-61) Knolls Atomic Power Lab., Schenectady, N.Y.
January 6, 1959. 46p. Contract W-31-109-Eng-52.

An analysis is presented which enables the design engineer to establish the proper amount of coolant flow to be bypassed into the control rod channel. Required inputs to the analysis are the heat generation, the total reactor flow, and the physical dimensions of the rod-channel combination. The analysis is extended to the specific case of the SAR configurations and the results presented in the form of curves which permit rapid evaluation of the thermal performance of the possible SAR rod-channel combinations. (auth)

- 5 - 2 BROOKHAVEN NATIONAL LABORATORY DESIGN MANUAL BOOK 1, VOL. 4. PILE AUXILIARIES

- - -
(M-4413) Ferguson (H. K.) Co., New York.
April 1949. Decl. January 13, 1956. 134p., For Brookhaven National Lab., Contract (AT-30-2-gen-16). (HKF-9).

This volume, No. 4 of the Pile Design Manual, consists of three chapters describing three units in the Pile Complex: the Fan House, Cooling Tower, and Stack. The location of these units is shown, and the functional relation is described. (auth)

- 5 - 3 COMPUTER ANALYSIS OF TRANSIENTS IN NUCLEAR REACTOR COOLANT NETWORKS
Korn, K. E., Jacobsen, R. C. and Woods, R. O.
(AECU-3866) Allstate Design and Development Co., Inc., Trenton.
June 4, 1958. 88p., For Combustion Engineering, Inc., Nuclear Div.
Contract AT(30-3)-198, Subcontract No. 66.

AECU-3867 supplements this report. Results are presented of a project to develop an engineering design tool in the form of a computer program which mathematically represents the transient flow behavior of the SIC nuclear power plant emergency cooling system. The computer program developed represents all the components of a nuclear reactor primary cooling system, including six possible flow paths which

are partially in parallel and partially coincident. As a design tool, the resultant computer program permits simulation of the transient flow behavior of a very large number of nuclear power plant coolant systems.

- 5 - 4 COMPUTER PROGRAM FOR TRANSIENTS IN NUCLEAR REACTOR COOLANT NETWORKS
Korn, K. E., Jacobsen, R. C. and Woods, R. O.
(AECU-3867) Allstate Design and Development Co., Inc., Trenton
June 4, 1958. 58p., For Combustion Engineering, Inc., Nuclear Div.
Contract AT(30-3)-198, Subcontract No. 66.

This report supplements AECU-3866. The program contains over 1300 equations, and was therefore written in Fortran for the IBM-704 computer. The program is described and listing given. The Fortran variables, subscript code, and nodal locations are described.

- 5 - 5 CONSIDERATIONS RELATIVE TO THE GAS FLOW CHANNELS AND HEAT TRANSFER SURFACES OF HIGH-TEMPERATURE OXIDE PILE
Robertson, A.
(N-2236) Chicago. Univ. Metallurgical Lab.
March 1, 1946. Decl. January 30, 1956. 26p.

Calculations are presented which confirm the feasibility of a He-cooled power reactor. Helium flow, friction losses, and pressure drop in the cooling system are shown. (D.E.B.)

- 5 - 6 CONTROL OF THE COOLING EQUIPMENT OF THE SACLAY REACTOR
Leduc, G.
L'Onde elec. 33:702-12. December 1953. (In French).

The properties of various possible gas refrigerants are reviewed and a detailed account is given of (a) the nitrogen/water system used for cooling the active uranium rods, (b) the air-cooling system for the tank containing the heavy water, (c) the arrangements for extracting the vapor above the surface of the heavy water in the tank, cooling it and returning it to the reservoir.

- 5 - 7 THE COOLANT CIRCUITS
Chaboseau, J. and Laferriere, P.
Bull. inform. sci. et. tech. No. 20, 89-98, August 1958. (In French)

A schematic diagram of the coolant circuits of the reactors is presented. The principal primary circuit and the auxiliary primary circuit are discussed in detail. The secondary circuit, compensation circuit, and the separation valves are considered more briefly. (J.S.R.)

5 - 8

COOLANT GAGS FOR THE CALDER REACTOR

Wilson, J. T. and Long, M.

(IGR-TN/W-319) Gt. Brit. Windscale Works, Sellafield, Cumb., England
March 28, 1956. 16p. (IGC-TRDC/P-161; IGC-TRDC/R-9).

Calder reactor coolant flow is regulated by means of discs or gags inserted in the bottom of the channels. Data are presented giving the relationship between gag size and pressure drop over a range of air flows for the three channel sizes in the Calder reactor. (auth)

5 - 9

CRITICAL SURVEY OF CHANNEL COOLING

Pulley, O.

(NURG/M-22) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.

June 26, 1952. Decl. November 21, 1956. 46p.

A detailed study was made of the corrections to theory required to give adequate accuracy for the design and performance of a single gas-cooled channel. For the most part, the corrections are unimportant in a natural U, graphite moderated pile with Al or Mg finning. Actual velocity, output and metal rating are almost, and the relative pressure drop completely, independent of the finning arrangements. The relative pressure drop is always small and the useful velocity is restricted to a range of less than 4 to 1. Conventional fin designs fix an upper limit to the outlet coolant temperature which is below that giving maximum output. The four coolants, air, CO₂, He and H are compared and an important difference found only in the case of H. (auth)

5 - 10

EQUALIZATION OF GAS TEMPERATURES ACROSS A HIGH TEMPERATURE OXIDE PILE BY REGULATING COOLANT GAS FLOW IN THE VARIOUS FUEL CHANNELS

Fromm, L. W., Jr.

(N-2300) Chicago. Univ. Metallurgical Lab.

June 6, 1946. Decl. January 27, 1956. 9p. (MUC-CAH-41)

It was previously found that the temperature of the exit coolant gases from the fuel channels of a pile will vary across the pile. The problem of equalization of these

temperatures is discussed, and several possible methods, all involving regulation of the gas flow in the channels, are described. These methods are: placement of orifices in channels in which gas flow is to be decreased; and division of gas stream and control of sections of the stream so that channels receive predetermined quantities of gas. The required increase in blower horsepower due to increased pressure drop across the pile is given as 23%. (auth)

5 - 11

ESTIMATES REGARDING GAS FLOW AND HEAT TRANSFER IN A HIGH TEMPERATURE OXIDE PILE

Robertson, A. F., Fromm, L. W. and Kittredge, H. E.

(CT-3460) Chicago. Univ. Metallurgical Lab.

March 1, 1946. Decl. December 20, 1955. 29p. Contract W-7401-eng-37.

Final design of the proposed BeO moderated pile, for operation at high temperatures to produce useful power, requires a choice of the optimal interrelated values of a number of values concerning pressure drops and heat transfer characteristics. To clarify the problems involved, estimates have been made of pressure drops for six different shapes of fuel rods; the effect of variation in alignment of the rods on the pressure drop; the fraction of pile power required for circulating the coolant gas; the effect of diameter of a hole in the fuel rods on gas flow distribution, total pile voids, and temperature distribution in gas streams and in the solid; the effect of assumed variation in heat production rates across the pile on the temperature difference between the fuel rods and coolant gas; the effect of the emissivity of BeO on the temperature difference between the fuel rods and the coolant gas; and the effect of pile power on temperature variation between coolant gas streams and walls of fuel tubes. (auth)

5 - 12

EXAMPLES FOR PRESSURE DROP CALCULATIONS IN PARALLEL FLOW HELIUM COOLING

Feld, B. T. and Szilard, L.

(GP-308) Chicago. Univ. Metallurgical Lab.

June 18, 1942. Decl. February 15, 1957. 4p. Contract (W-740-eng-37). (A-350).

Pressure drop calculations are shown for He-cooled power plants in the range 30 to 400 Mw. He pressures of 1 and 10 atm. are used. (D.E.B.)

5 - 13

GAS-CIRCULATION PROBLEMS IN REACTOR DEVELOPMENT
Ridley, J. R.
G.E.C. Atomic Energy Rev. 1, 160-5, March 1958.

A brief account is given of the arrangement of the gas-circulator system for the Hunterston reactors, and of the considerations that led to that particular design. Experimental work on a one-third-scale model of the blower is described, and a typical characteristic illustrated. Different types of shaft seal are discussed, and experiments with two of them--the carbon face seal and the viscous oil seal--are described. (auth)

5 - 14

GAS-COOLED ATOMIC REACTOR EQUIPMENT
- - -, (to Allmanna Svenska Electriska AB (Sweden).
British Patent 775, 602. Nuclear Eng. 2, 452(1957) October.

Gases, preferably CO₂ compressed to about 60 atm., are especially suitable as heat transmission media. There are, however, some constructional problems connected with the use of highly compressed gases. The reactor consists of an Al tank filled with heavy water and surrounded by a graphite mantle as neutron reflector. Fuel rods of U or Th are inserted in the tank and are, e.g., hollow or H-shaped so that highly compressed gas can flow along these rods for cooling and heat transmission. Distance members of Zr or Be are inserted between the rods. They are surrounded by tubes of a material similar to that of the distance members. The tubes are fixed in a gas distributing and a gas collecting vessel with numerous inlet and outlet pipes. In order to reduce the pumping power required for the gas at such high pressure, the collecting vessel is divided into a number of ring-shaped chambers. The pipes are connected to individual pump systems and the arrangement permits equalizing of the temperature by throttling the cross-section of the flow of gas. (auth)

5 - 15

THE HANDLING OF RADIOACTIVE-CONTAMINATED AIR AT OAK RIDGE NATIONAL LABORATORY
Stockdale, W. G., Oak Ridge National Lab., Tenn.
In Third Atomic Energy Commission Air Cleaning Conference Held At Los Alamos Scientific Laboratory, (WASH-170(Del.)), September 21, 22, and 23, 1953. p. 21-35.

The systems in use at Oak Ridge National Laboratory for decontamination of radioactive-contaminated air and its final disposal are described. The air-cleaning facilities for the ORNL Graphite Reactor, laboratories and chemical plants, radiochemical process vessel off-gas, and Homogeneous Reactor Experiment are described. (W.L.H.)

5 - 16

HEAT TRANSFER ARRANGEMENTS

Boyland, D. A. (to General Electric Co., Ltd.)

British Patent 790, 688. Nuclear Eng. 3, 414 September 1958.

The difficulties of gas-cooling are overcome by the arrangement of the path of the coolant gas transversely to the fuel elements, e.g., by ducts for the flow of coolant on each of two opposite sides of the fuel elements, one serving as the inlet, the other as an outlet, so that the coolant flows across between the transverse cooling fins. The flow may be controlled by decreasing the cross-section of the inlet duct in the direction of flow, and at the same time increasing that of the outlet.

5 - 17

IMPROVEMENTS IN NUCLEAR REACTORS AND AN IMPROVED METHOD OF OPERATING NUCLEAR REACTORS

- - -, (to Babcock & Wilcox, Ltd.)

British Patent 794,329. April 30, 1958. Nuclear Power 3, 348(1958) July.

Instead of preventing the circulation of gaseous fission products in a gas-cooled reactor by canning the fuel elements, this is an idea for a reactor in which the poisons, notably xenon, are removed from the coolant chemically. Unsheathed fuel rods are inserted into vertical channels in the graphite moderator and pressurized carbon dioxide passed upwards over them in the usual way. The horizontal boiler has hemispherical ends and is divided by a partition so that the coolant flows in a U-shaped path. Part of the coolant is led off to a water-filled pressure vessel where the carbon-dioxide dissolves and the insoluble xenon with some CO₂ is blown off. The water is then passed through a throttling valve into another vessel with lower internal pressure, and degasification takes place. Liberated CO₂ is returned by a compressor to the closed circuit passing through a freezer to remove water vapor. Only part of the coolant need be purified as a certain enrichment in xenon is permissible.

5 - 18

IMPROVEMENTS IN OR RELATING TO NUCLEAR REACTOR ARRANGEMENTS

Lindley, P. A. (to General Electric Co., Ltd.).

British Patent 789,482. January 22, 1958. Nuclear Power 3, 200 April 1958.

Hunterston is the only one of the new British nuclear power stations that is using centrifugal blowers for gas circulation, and the reasons for General Electric's choice are put forward in this patent. Owing to the relatively small pressure rise needed across the blower in these plants only one stage is needed and the centrifugal type would seem to be

very suitable. The static pressure is, however, very high and the normal type of diffuser scroll casing is considered fundamentally unsuitable to withstand this. The designers have therefore proposed to get over this difficulty, while retaining a centrifugal impeller, by installing the blower actually inside the boiler. By this means, the diffuser casing need only be capable of withstanding the differential pressure and can therefore be made in comparatively light material. The impeller and diffuser sections can be very large, thus contributing to efficiency. The invention also covers axial- and mixed-flow impellers.

5 - 19

IMPROVEMENTS IN OR RELATING TO NUCLEAR REACTORS

Fortescue, P. and Middleton, J. P. (to United Kingdom Atomic Energy Authority).

British Patent 799,212. August 6, 1958. Nuclear Power 3, 524 October 1958.

Considerable saving in pumping power in the gas circulation circuit is achieved by directly linking a gas turbine to the compressor and incorporating it in the coolant circuit. In principle the whole of the heat output is passed on to the secondary heat exchanger, but in fact there will be a small temperature drop across the turbine. It is particularly applicable to gas-cooled, solid homogeneous or gaseous-fuelled reactors. In the design described, primary coolant is helium pressurized to 10 atm, inlet and outlet temperatures are 350 and 750°C, and the heat output is approximately 30 Mw. Turbine inlet and outlet pressures are 142.4 and 117.8 psi, and the compressor has a compression ratio of 1.2771 with an inlet pressure of 115.5 psi.

5 - 20

IMPROVEMENTS IN OR RELATING TO NUCLEAR REACTORS

le Foll, J. and Melese, G. (to Commissariat a l'Energie Atomique).

British Patent 792,171. March 19, 1958. Nuclear Power 3, 254 May 1958.

In a gas-cooled reactor the heat output is proportional to the neutron flux and is therefore highest near the center of the core, decreasing radially outwards to the periphery. Higher rated channels are limited by the maximum temperature in the uranium, and in peripheral channels the limit is set by the maximum permissible can temperature. In an intermediate zone both factors have to be taken into account. There is an optimum thermodynamic condition for each channel and to obtain this, dividing the channels into annular groups is proposed. In a particular application—which seems to be similar to the arrangement used in the Marcoule reactors—there are three annular inlet zones supplied from concentric manifolds. Temperatures are 50, 150, and 200°C

reading radially outwards. Inlet pressure is the same in each (200 psi). On the outlet side there are two annular manifolds to each inlet one and the flows are mixed by jet pumps.

- 5 - 21 INVESTIGATION OF PRESSURE DROPS CAUSED BY URANIUM PLATES IN THE HELIUM COOLED PLANT
David, M. M., Jr.
(CE-481) Chicago. Univ. Metallurgical Lab.
February 22, 1943. Decl. January 18, 1956. 23p. Contract (W-7401-eng.-37) (A-470).

Full scale wooden replicas were used in an experimental study of the pressure drop caused by the flat lugged plate assemblies proposed for one design of the helium cooled power plant. Air at about 1 atm. was used instead of the helium which would be at 7 atm. in the plant, since the densities are the same within a factor of two or less. The air velocities ranged up to 250 ft/sec at about 120 ft/sec, which approximates a design condition, the pressure drop per element ranged from 0.077 psi to 0.30 psi depending on the plate thickness and plate edge treatment. Three plate thicknesses, 3/16 in., 1/4 in. and 3/8 in. were tested, and various degrees of tapering of the edges were tried. The best results came from tapering both the leading and trailing edges to a 30° included angle (the smallest tried). This treatment reduced entrance-exit losses from about 2.0 velocity heads to about 0.5 velocity heads per element. Friction losses (calculated) ran from about 0.17 to 0.31 velocity heads. Tapering of the edges of the plates is evidently essential if this design is to be used.
(auth)

- 5 - 22 LARGE SCALE EXPERIMENTAL STUDY OF PRESSURE LOSS IN AIR FEED SLOT
Beiley, I.
(BNL-3110) Ferguson (H.K.) Co., New York
March 8, 1948. Decl. June 8, 1956. 58p. (For Brookhaven National Lab.).

The results of the second experimental gap study on a large-scale model at M.I.T. are given. These data are necessary for computation of channel fore-pressures in the Brookhaven Reactor at various air flow rates. (T.R.H.)

5 - 23

LEICHTWASSER - ODER HOCHDRUCKGASKÜHLUNG BEI EINEM 100-MW-
REAKTORPROJEKT

Kornbuchler, H.

Atom u Strom Vol. 4, No. 9, September. p. 71-6 (Supp to Elektrizit-
ätswirtschaft Vol. 57 No. 17, September 5, 1958).

Water vs high pressure gas cooling in 100-Mw reactor project; studies by AEG on water cooled reactor, also considering use of high pressure gas; combination of natural uranium fuel and heavy water moderator, and slightly enriched uranium fuel and graphite moderator will give about same price per kw-hr, lower than that for water-cooled reactor; reduction in cost of water cooled reactor may be compensating factor.

5 - 24

A MOLTEN SALT NATURAL CONVECTION REACTOR SYSTEM

Romie, F. E. and Kinyon, B. W.

(CF-58-2-46) Oak Ridge National Lab., Tenn.

February 5, 1958. 27p. Contract (W-7405-eng-26).

Fuel-salt volumes external to the core of a molten-salt reactor are calculated for a system in which the fuel salt circulates through the core and primary exchanger by free convection. In the calculation of these volumes, the exchanger heights above the core top range from 5 to 20 ft. Coolants considered for the primary exchanger are a second molten salt and helium. External fuel holdup is found to be the same with either coolant. Two sets of terminal temperatures are selected for the helium. The first combination permits steam generation at 850 psia, 900°F. The second set is selected for a closed gas turbine cycle with an 1100°F turbine inlet temperature. Specific power (thermal kw/kg 235) is found to be about 900 kw/kg, based on initial, clean conditions and a 60 Mw (thermal) output. A specific power of 1275 kw/kg is estimated for a forced convection system of the same rating. (auth)

5 - 25

NUCLEAR REACTOR ARRANGEMENTS

Lindley, P. A. (to General Electric Co., Ltd.).

British Patent 789,482. Nuclear Eng. 3, 360 August 1958.

A gaseous cooling medium is usually circulated through the reactor core and the heat exchange system at relatively high pressure. If a centrifugal impeller is used, it is surrounded by a diffusing casing, and the required shape, large in diameter but of short axial length, is unsuited to withstand high internal pressures. The problem has been solved by arranging both impeller and diffusing casing within the existing pressure vessel surrounding a heat exchanger or the reactor core.

5 - 26

THE PERFORMANCE OF A CO₂-COOLED PILE IN A PRESSURE ENVELOPE. HEAT TRANSFER AND PRESSURE DROP CALCULATIONS

Hodge, J. and Smith, J.

(FPP-12) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.

May 6, 1948. 21p.

General equations are developed for estimation of the heat transfer and pressure drop in the coolant channels of a pile with finned slugs, using CO₂ at an inlet pressure of 5 atmospheres absolute as coolant. For various assumptions of values of pile length, coolant channel area, fin length, coolant gas inlet temperature, and coolant flow velocity, values of coolant outlet temperature and mass flow, maximum permissible rating of the channel and pumping power (as a fraction of output) required to overcome the pressure drop in the coolant channel only are obtained for each of the cases; the maximum sheath temperature was fixed at 400°C throughout. The detailed results are given. Interpolation for outlet temperatures of 275, 300 and 325°C, and the omission of cases where either outlet temperature is too low or pumping power too high, leads to the values given. Since the calculations were made, further experimental data on heat transfer have indicated a fin efficiency below that used in the report, so an addendum was prepared to demonstrate the effect of reducing this efficiency to 0.80 of the original value on a section of the results of which the typical value above is one. For given parameter values, the permissible Mw/ton in the string is reduced by about 10% and the pumping power increased by about the same amount. (auth)

5 - 27

POWER REACTOR STUDIES

Isakoff, L., comp.

(DP-265) Du Pont de Nemours (E.I.) & Co. Atomic Energy Div., Wilmington, Del.

Quarterly Progress Report (for) August, September and October 1957. comp. February 1958. 79p. Contract AT(07-2)-1.

An evaluation is made of gas cooling versus heavy water cooling for power reactors that are fueled with natural uranium metal and moderated with heavy water. The power reactors are designed to produce 100 Mwe. Basic physics data, engineering design problems, metallurgical developments, and estimates of the costs associated with the power reactors also are given. (For preceding period see DP-245.) (auth)

5 - 28

A PRELIMINARY EVALUATION OF GAS COOLING OF POWER REACTORS MODERATED BY HEAVY WATER

Holmes, R. C., Bernath, L. and Hones, E. W.
(DP-307) Du Pont de Nemours (E. I.) & Co. Atomic Energy Div.,
Wilmington, Del.
August 1958. 50p. Contract AT(07-2)-1.

A preliminary study of gas-cooled natural uranium power reactors moderated by heavy water and a comparison of them with a similar reactor cooled by heavy water have shown that a reactor cooled by helium holds sufficient economic promise to merit further study. The results indicate that helium cooling might give a lower cost for power by about 10%. However, many other developments must be made in the design of reactors in order to reduce the cost of electrical power from atomic energy to an economical zone of 4 to 6 mills per kwh. The data were generated for the purpose of guiding the over-all research program relating to the production of low cost power from heavy-water-moderated atomic reactors. (auth)

5 - 29

QUARTER-POINT GAS INTRODUCTION SCHEME FOR HELIUM COOLED PLANT

Leverett, M. C.
(CE-288) Clinton Lab., Oak Ridge, Tenn.
October 6, 1952. Decl. December 13, 1955. 9p. (A-308)

The engineering problems involved in a proposal to alter the helium cooled pile design so as to introduce the cold helium at the top, bottom and middle of the pile, and to remove hot helium from quarter-points of the pile are discussed. The main reason for this procedure is that it might give a lower average temperature of the pile than the straight series flow. The conclusion of the study is that the temperature would not be lowered substantially, and that the only other considerable advantage is the reduced pressure drop. This can be obtained with less complexity by parallel flow and the quarter-point scheme is therefore not recommended. (auth)

5 - 30

SOME CHARACTERISTICS OF REACTOR GAS COOLING SYSTEMS

Fortescue, P. (A.E.R.E., Harwell).
Nuclear Power 2, 188-92(1957) May.

With gas-cooled reactor fuel cans using circumferential fins or grossly roughened surface, heat transfer work with a given fin material and coolant gas can be generalized by plotting Stanton number against Reynolds number multiplied by the ratio of the thermal conductivity of the gas to that of the metal. For this type of surface a figure of merit in terms of molecular weight and specific heat is obtained. (auth)

5 - 31

THERMAL NUCLEAR REACTOR

Fenning, F. W. and Jackson, R. F. (to U.S. Atomic Energy Commission).

U. S. Patent 2,807,580. September 24, 1957.

Nuclear reactors of the graphite moderated air cooled type in which canned slugs or rods of fissile material are employed are discussed. Such a reactor may be provided with a means for detecting dust particles in the exhausted air. The means employed are lengths of dust absorbent cord suspended in vertical holes in the shielding structure above each vertical coolant flow channel to hang in the path of the cooling air issuing from the channels, and associated spindles and drive motors for hauling the cords past detectors, such as Geiger counters, for inspecting the cords periodically. This design also enables detecting the individual channel in which a fault condition may have occurred.

5 - 32

TREATMENT OF THE FLUID

Dequenes, P. and Garnier, M.

Bull. inform. sci. et tech. No. 20, p. 115-25, August 1958.

(In French)

The carbon dioxide is treated before its introduction into the reactor, during its storage, just before feeding into the reactor during its circulation by continuous purification, and at the moment of its evacuation to the atmosphere by venting. The methods used at each stage are described.
(J.S.R.)

6. CORE STRUCTURE

6 - 1 ANALYSIS AND RECOMMENDED MODIFICATIONS FOR DUCTS AND PLENUM CHAMBERS AT BROOKHAVEN NATIONAL LABORATORY

Boone, M. and Schoessow, E. E.

(M-4596) Babcock and Wilcox Co., New York

January 24, 1950. Decl. February 2, 1956. 163p., For the H. K. Ferguson Co., Contract AT-30-2-gen-16, Subcontracts S-26 and PRO-111 (S26-146).

Recommendations are presented for reinforcing and altering the design of air ducts and plenum chambers of the Brookhaven Reactor. It was concluded that these changes would raise the strength level and increase the service life of the structure. Drawings and supporting calculations for the recommendations are included. (C.H.)

6 - 2 ANALYSIS AND RECOMMENDED MODIFICATIONS FOR PILE STRUCTURE AT THE BROOKHAVEN NATIONAL LABORATORY

- - -

(M-4575) Babcock and Wilcox Co., New York

June 15, 1950. 554p., For H. K. Ferguson Co., Contract AT-30-2-GEN-16. Subcontracts S-26 and PRO-111-(S-26-146).

Recommendations are presented for altering the design of the Brookhaven Reactor to raise the strength level and increase the service life of the pile structure. Drawings and supporting calculations for the recommendations are included. (C.H.)

6 - 3 THE CORE OF THE G-2 AND G-3 REACTORS

Deleuze, G.

Bull. inform. sci. et tech. No. 20, p. 36-55, August 1958.

(In French).

All parts of the reactor contained within the concrete shield are described in detail. The graphite block, the wall of graphite protecting the rear face, the thermal shielding screen, the graphite supports, the bands for the graphite on its side and upper faces, the pressure supports of the graphite, the loading unit, the channels for fuel element discharge, the control and safety rods, the compartmentation of the coolant gas, and the installation

of detection and measurement instruments are discussed. The protection of the metallic parts of the core from corrosion is briefly considered. (J.S.R.)

- 6 - 4 IMPROVEMENTS IN OR RELATING TO FUEL ELEMENTS FOR NUCLEAR REACTORS
Dent, K. H. (to United Kingdom Atomic Energy Authority).
British Patent 784,890. August 25, 1955. Nuclear Power 3, 92(1958) February.

At one time it was proposed to use separate supports for the Calder Hall fuel elements and this patent covers the idea. The idea was to attach a spider with three hinged legs to the top of each fuel cartridge. The legs were spring-loaded outwards and located in circumferential grooves cut in the graphite blocks. Since neutron economy is so important, these spiders would have to be in beryllium or zirconium. A beryllium one was illustrated at the BNEC's Calder Hall symposium.

- 6 - 5 IMPROVEMENTS IN OR RELATING TO SUPPORT DEVICES FOR USE IN NUCLEAR REACTORS
Greenhalgh, F. C. and Long, E. (to United Kingdom Atomic Energy Authority).
British Patent 783,094. July 28, 1954. Nuclear Power 3, 92(1958) February.

Another Calder patent, this deals with the method of supporting the stack of fuel elements to allow for differential expansion between the base support members and the moderator. It also provides a shear pin and broach device at the bottom of the column so that the accidental dropping of a fuel element on to the column does not cause permanent damage. The assembly also includes an annular plate, known as the gag, by means of which the amount of coolant entering the particular channel can be regulated.

- 6 - 6 NUCLEAR REACTOR WITH REPLACEMENT INSTALLATIONS BETWEEN THE FUEL LATTICE
- - - (To Siemens-Schuckertwerke AG., Berlin u. Erlangen, Ger.)
German Patent DAS 1033 808. Atompraxis 5, 43(1959). (In German).

In a gas-cooled or boiling water reactor with rod lattice whose fuel rods are hollow cylinders and are in contact on the inside as well as on the outside with a flowing coolant, replacement installations which are massive or hollow,

cylindrical, or prismatic bodies are arranged in the gaps between the rod lattice. In this manner the proportional external diameter of each fuel element or the corresponding volume was determined from the corresponding measurements of the internal flow channel.

6 - 7

NUCLEAR REACTORS

Dennis, W. E and Lindley, P. A. (to General Electric Co., Ltd.).
British Patent 791,011. Nuclear Eng. 3, 414(1958) September.

In gas-cooled nuclear reactors the avoidance of mechanical abrasion in the fuel channels of the reactor core during charging and discharging operations is one of the problems to be solved. Other problems are to avoid chemical attack on the moderator by the gas with or without mass transfer of moderating material, and to overcome the effects of radioactive contamination which would render the channel unsuitable for further use. The problems were solved by housing one or more fuel elements, within a tubular member constituting a lining or part lining for a channel, and forming, together with the fuel element, a replaceable unit. The gas then flows through the tubular member to abstract the heat from the fuel elements which are spaced away from the inside wall of the member which may be made of moderating material.

6 - 8

NUCLEAR REACTORS

- - - (to Stichting Reactor Centrum Nederland (Netherlands)).
British Patent 788,284. Nuclear Eng. 3, 314(1958) July.

In the reactor, one or more bodies of porous material are provided in the reaction zone consisting of an intimate mixture of fuel, moderator, and heat distributing material. The fissile material is not present as metal but is distributed throughout the porous material formed by a sintering process. The gaseous fluid is pumped through the reaction zone under pressure. Core temperatures of 600 to 1500°C are attainable. (W.D.M.)

6 - 9

REACTOR CORE STRUCTURE OF THE PEGCPR

Rulien, F. R., (Nuclear Products-Erco Div.), ACF Industries, Inc.
Washington, D. C.

In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564(p.63-9)). October 21-22, 1958.

The graphite core structure is described, and the bases for the selection of the more important materials are discussed. The operating conditions are given and the effects of these conditions on the design of the core structure are reviewed.
(auth)

7. CORROSION

7 - 1 CORROSION PROBLEMS IN GAS-COOLED REACTORS

Darras, R.

Bull. inform. sci. et tech. No. 21, p. 13-25, September 1958.
(In French).

The corrosion behavior of different materials used in gas-cooled reactors is reviewed. A brief report is made on the dry oxidation of metals and the methods of investigation. The materials studies are the nuclear fuels (uranium and its compounds), cladding materials (magnesium, aluminum, zirconium, and beryllium and their alloys and stainless steel), and moderators (graphite and beryllium oxide). An attempt is made to specify the temperature limits in each material for various gases. The possibilities of improving the energy yield of gas cooled reactors by the use of more refractory materials or the use of other gases is discussed. 38 references. (auth)

7 - 2 SOME CORROSION CONSIDERATIONS OF HIGH TEMPERATURE GAS-COOLED REACTORS

Yoslm, S. J., McKisson, R. L., Saul, A. M. and McKenzie, D. E.
Atomics International Div., North American Aviation, Inc., Canoga Park, Calif. A/CONF.15/P/1077. 11p.

The basic chemical principles by which the corrosion properties and thus the temperature limitations of refractory materials may be evaluated are described. High-temperature corrosion by direct vaporization and the reaction of a gaseous product are considered. (W.D.M.)

8. DESIGN

- 8 - 1 AN ADDENDUM TO A PARAMETRIC STUDY OF THE GAS-COOLED REACTOR CONCEPT
 - - -
 (HW-54727(Add.)) (Rev.) General Electric Co. Hanford Atomic Products
 Operation, Richland, Wash.
 March 1, 1958. 251p. Contract W-31-109-Eng-52.

This addendum to the report of parametric studies on the gas-cooled reactor concept was prepared to serve a twofold purpose: to provide background information regarding the study results presented in the first volume and to provide additional information resulting from the studies made on the gas-cooled reactor concept which, though not directly applicable to the cases studied, may be of value in the evaluation of other studies of the gas-cooled reactors. Studies are presented on fuel elements, reactor physics, heat transfer and fluid flow, coolant gas selection, core arrangement and vessels, direct and indirect cycle power plants, and costs. (W.D.M.)

- 8 - 2 ADVANCED DESIGN STUDIES
 Perry, A. M. Oak Ridge National Lab., Tenn.
 In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge
 National Laboratory, (TID-7564(p.334-46)). October 21-22, 1958.

The purpose of advanced design studies, at the present stage of gas-cooled reactor development, is mainly to disclose those combinations of materials and operating parameters that show the greatest promise of yielding low cost electric power or of satisfying some other design criterion such as successful breeding. Preoccupation with the reactor now under development has been such that not a great deal of advanced design work has yet been undertaken. A few rather well defined studies have been carried out: an investigation of the breeding capabilities of graphite-moderated, gas-cooled reactors, a brief study of high-pressure systems, and a nuclear gas engine for maritime applications. (auth)

- 8 - 3 APPENDIX TO PRELIMINARY DESIGN, 55,000 KW PROTOTYPE NATURAL URANIUM,
 GAS COOLED, GRAPHITE MODERATED NUCLEAR POWER PLANT (PROTOTYPE FOR AN
 OPTIMUM PLANT). (Cover carries title: APPENDIX TO GAS COOLED POWER
 REACTOR, PRELIMINARY DESIGN, 55,000 KW PROTOTYPE NATURAL URANIUM
 NUCLEAR POWER PLANT).

- - -
(IDO-2021) Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif.
and Nuclear Products-Erco Div., ACF Industries, Inc., Washington, D.C.
March 1958. Revision 1, April 1, 1958. 136p. Contract AT(10-1)-925.
58-1-RE (Rev. 1, App.)

Outline specifications are given in the following categories:
civil, architectural; structural; heating, ventilating, and
air conditioning; piping systems and associated equipment;
electrical; mechanical; instrumentation; and reactor components.
An outline of startup, operation, and maintenance procedure,
the preliminary estimate of cost, the construction schedule,
and the preliminary reactor safety analysis is included. Data
for the comparison of longitudinal, transverse, and helical
finned fuel elements are presented. Drawings are given for
all phases of the plant included in the outline specifications.
(M.H.R.)

8 - 4 BROOKHAVEN NATIONAL LABORATORY DESIGN MANUAL. BOOK NO. 1. VOLUME 3.
PARTS 1-111. PILE

- - -
(M-4412) Ferguson (H.K.) Co., New York
April 1949. Decl. February 15, 1956. 330p. (HKF-8)

The nuclear facilities at BNL are described. Fundamental
considerations influencing the design are reviewed. The
fundamental design data set up for each unit, the basic
data and assumptions used in carrying out the design, and
a record of the completed design are presented. Drawings
and photographs are included. (C.H.)

8 - 5 CALCULATION OF THE OPTIMUM COOLANT STATE IN GAS COOLED REACTORS
Benzler, H. (Babcock und Wilcox, Oberhausen, Germany)
Atomkern-Energie 3, 331-6(1958) August - September. (In German).

Net power and thermal efficiency of a reactor plant can be
calculated by two simple equations as functions of coolant
pressure drop and reactor inlet and outlet temperatures.
There exists a characteristic curve in the power-efficiency
diagram, all points of which define the most favorable cool-
ant design values. The construction of the curve is de-
scribed. (auth)

8 - 6

CALDER REACTORS: RESEARCH AND DEVELOPMENT PROBLEMS

Rotherham, L.

Nature (Lond), Vol. 178, No. 4538, p. 825-8, October 20, 1956.

Problems in design of reactors at Calder Hall Atomic Power Station; reasons for choice of graphite moderated reactor cooled with carbon dioxide; methods used in accurately estimating reactivity of fully loaded core by exponential experiments; research work to improve heat transfer of fuel; problems relating to pressure vessels, fuel elements, etc.

8 - 7

CHAPEL CROSS: BRITAIN'S SECOND FULL-SCALE NUCLEAR POWER STATION
OPENED NEAR ANNAN, DUMFRIESSHIRE

- - -

Nuclear Eng. 4:250-2(1959) June.

Construction procedure and design changes are described. The changes in design from that of Calder Hall concern the fuel channels, the pressure vessel, the coolant circuit, and a cooling pond. (W.H.R.)

8 - 8

CLASSIFIED PROGRESS REPORT OF THE REACTOR SCIENCE AND ENGINEERING
DEPARTMENT FOR 1949

- - -

(BNL-48) Brookhaven National Lab., Upton, N.Y.
April 1950. Decl. February 15, 1957. 107p.

Efforts during the report period have been divided between two fields; research, and evaluation and redesign of the reactor. Greater effort has been directed toward re-evaluation of the structural design of the reactor, tests on various components, and engineering design aimed at correcting such structural defects as have been uncovered. Along with these have been included studies for the start-up procedure and actual testing and readying of the reactor auxiliary equipment and instrumentation. Progress has been made on certain aspects of the research program, as well. (auth)

8 - 9

COMMONWEALTH EDISON COMPANY--PUBLIC SERVICE COMPANY REPORT ON POWER
GENERATION USING NUCLEAR ENERGY

Evers, L. A. and Kiefer, W. M. Commonwealth Edison., Chicago.
In Nuclear Science and Technology, (TID-2503(Del.)) p. 139-80
1958.

This article is a condensation of CEPS-1101.

Two conceptual natural-U reactor designs are offered: a

350-Mwh, 45-Mwe, helium--cooled, graphite-moderated; and a 1064-Mwh, 211.5-Mwe, D₂O-moderated and D₂O cooled. The total estimated costs are 40 and 118 million dollars, respectively. (T.R.H.)

8 - 10 COMPARISON OF INDUSTRIAL DESIGNS

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Nuclear Eng. Vol. 2, No. 10, January 1957. p. 5-9.

Nuclear power station designs prepared by four groups of companies for inclusion in their bids are compared; they concern erection of generating stations at Bradwell in Essex, Berkeley in Gloucestershire and Hunterston in Ayrshire; designs are based on Calder Hall plant as prototype, but in many points of detail they differ substantially; large advance in power output offered.

8 - 11 COMPARISON OF OPTIMUM NATURAL URANIUM NUCLEAR POWER PLANT AND OPTIMUM ENRICHED URANIUM NUCLEAR POWER PLANT. (GAS-COOLED POWER REACTORS).

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(IDO-24026) Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif. and Nuclear Products-Erco Div., ACF Industries, Inc., Washington, D. C.
August 1958. 237p., Contract AT(10-1)-925. 58-19-RE.

The designs of an optimum natural uranium power plant and an optimum partially enriched uranium power plant, both using CO₂ as a coolant, are compared. The basis for optimization of both plants is set forth and a description of the design of each plant is included. Those features of design, construction, and economic considerations which differ between the two optimum plants are discussed in detail. The design of each of the plants incorporates new advances in technology and the best features of similar plants now operating, or under construction, to provide net plant efficiency higher than has thus far been achieved in nuclear plants. (auth)

8 - 12 A CONCEPTUAL STUDY OF A 100 KW GAS COOLED REACTOR POWER PLANT FOR A REMOTE LOCATION

Frankfort, J. H.
(WKNL-46) Kidde (Walter) Nuclear Labs., Inc., Garden City, New York.
May 31, 1955. Decl. with deletions May 7, 1957. 179p. Contract AT(11-1)-339.

An analysis and design study is presented for a gas-cooled reactor suitable for supplying 100 kw of electrical energy and 400 kw of heat to a distant early warning radar station.

The graphite reflector-moderated reactor with UO_2 -stainless steel fuel elements is helium cooled. The helium is also the working fluid for a closed cycle regenerative gas turbine power plant. The plant, which requires only one operator at the console at all times, is designed for packaging in the continental United States, shipment by air to a remote base, and erection on a permafrost site during the four-month construction season. (J.S.R.)

8 - 13

DESCRIPTION OF A NATURAL URANIUM REACTOR SUPPLYING HIGH TEMPERATURE HEAT

Clark, F. H. and Klahr, C. N.
(NP-7303) Technical Research Group, New York.
July 1957. 30p.

A natural uranium reactor is proposed which is designed to combine the advantages of a natural uranium system with high generating system thermal efficiencies made possible by superheat from the reactor. The reactor is to have two power producing zones having different composition and properties. One zone, operated at relatively low temperature, affords sufficient reactivity for the entire system to be fueled with natural uranium. The other zone contains high-temperature materials and is the source of superheat supply. Computations establish that the proper balance of power from the two zones required for a practical generating system can readily be achieved in reactor systems presently feasible. Two systems are dealt with in this investigation: (1) a sodium-gas system which is based entirely upon developed technology and for which detailed calculations are presented and (2) an all gas system which requires further development. (A.C.)

8 - 14

DESIGN CONCEPT FOR A GAS-COOLED, PRESSURE TUBE D_2O -MODERATED REACTOR
Roy, G. M. and Valerino, M. F., General Nuclear Engineering Corp.,
Dunedin, Fla.

In Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 216-33.

The design concept for the East Central Nuclear Group-Florida West Coast Nuclear Group gas-cooled project includes pressure tubes, CO_2 coolant, and D_2O moderator. The attainment of high gas temperature permits the utilization of an efficient steam cycle. Net electrical output of the prototype is to be 50 Mw; of the large scale reactor, 200 Mw. The gas coolant temperature is to be at least 1000°F . (W.D.M.)

- 8 - 15 DESIGN FOR A DUAL PURPOSE REACTOR (G2)
Ailleret, P. P., Taranger, P. and Yvon, J.
In International Conference on the Peaceful Uses of Atomic Energy.
Geneva, 1955, Proceedings. New York, United Nations, 1956. Power
Reactors, Vol. 3, p. 84-85.

Description of a reactor using natural uranium as a fuel, graphite as the moderator, and compressed carbon dioxide as the fuel. The arrangement of the core, the method of renewing fuel elements, and the cooling circuit are discussed. The pressure vessel is of prestressed concrete with an inner lining of steel. The concrete is kept cool by circulating cold carbon dioxide between it and the liner.

- 8 - 16 DESIGN OF AN EXPERIMENTAL REACTOR FOR HIGH OPERATING TEMPERATURES
Davidson, L., DeFelice, J. and Klapper, J. A.
(NDA-64-101) Nuclear Development Corp. of America, White Plains, N.Y.
September 28, 1956. 141p., Contract AT(30-1)-862.

A preliminary design of a nuclear reactor to operate at 2500°F is described. The reactor is a prototype of a process heat reactor. There are two power producing regions in the reactor: an inner, He-cooled 2500°F region and an outer, water-cooled low temperature region. (auth)

- 8 - 17 DESIGN PARAMETERS IN THE ORNL GAS-COOLED REACTOR STUDY
- - -
Nuclear Eng. 3, 282-5(1958) July.

The design parameters and economics of the enriched uranium oxide, stainless steel clad, helium-cooled reactor designed by ORNL are reviewed. (W.D.M.)

- 8 - 18 DESIGN STUDY FOR A GRAPHITE-MODERATED GAS-COOLED REACTOR USING
PARTIALLY ENRICHED URANIUM
Charpie, R. A., et al., Oak Ridge National Lab., Tenn.
A/CONF.15/P/450, et al. 16p.

A summary is presented of the research and development aspects of the U. S. program for graphite-moderated, gas-cooled reactors using partially enriched uranium. The availability of helium and enriched uranium in the United States presents a large degree of design freedom. On the basis of the study, it is concluded that such a reactor will produce power cheaper than a reactor using natural uranium. At the present time gas-cooled reactors are technologically and economically competitive with the best available pressurized-water reactors. (J.S.R.)

8 - 19

A DESIGN STUDY OF TWO HELIUM-COOLED REACTORS

Armstrong, R. H., Lovoff, A. and Templin, L. J., Argonne National Lab., Lemont, Ill.

In Nuclear Science and Technology, (TID-2508) (Del.), December 1955. p. 445-61.

A conceptual design for two reactors, one of 500-Mw and the other of 189-Mw heat output, is presented. The reactors are cooled with helium so that they can be matched to an external closed-cycle gas-turbine system. The helium operates at a pressure of 100 psi and a reactor outlet temperature of 1400°F. Because of the high temperatures, graphite moderator and thorium fuel are considered. The 500-Mw reactor operates with 17.5 tons of Th containing 3.61 wt. % U^{233} , the 189-Mw reactor operates with 6.2 tons of Th containing 3.47 wt. % U^{233} . Both reactors have initial conversion ratios of approximately 0.92 and require an indeterminate extra enrichment for long-term operation. The primary design limit chosen is that of a maximum heat flux of 291,000 Btu/hr/sq. ft. Materials temperature limitations are roughly compatible with this value. (auth)

8 - 20

DEVELOPMENT OF GAS-COOLED REACTORS FOR POWER PRODUCTION

Moore, R. V.

Engineer, Vol. 202, No. 5265, p 881-3, December 21, 1956.

Effect of fuel temperatures and coolant conditions on performance, bearing in mind possibilities of dispensing with steam cycle and using coolant in gas turbine; prospects of fuel recycling as means of reducing running costs of nuclear power generation. From paper presented at symposium on Calder Works Nuclear Power Plant.

8 - 21

DEVELOPMENT OF GAS-COOLED REACTORS FOR POWER PRODUCTION

Moore, R. V.

Instn. Elec. Engrs.--Proc., Vol. 104, pt A (Power Eng.), No. 16, p. 281-8, August 1957.

Gas cooled, graphite moderated reactor is capable of considerable development; larger reactors of Calder Hall type will produce electric power at competitive cost in United Kingdom; promising line of development towards higher temperature operation requires development of new materials, particularly cladding of uranium; use of oxide and carbide forms of uranium and recycled plutonium as fuel. Paper 2363.

8 - 22 THE DRAGON H.T.G.R. PROJECT, A PRELIMINARY DESIGN STUDY

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Atomic World 10, 178-9, 185(1959) May.

Discusses preliminary designs and lists the most important problems to be investigated in the Dragon high-temperature gas-cooled (He) reactor experiment to be constructed at Winfrith Heath. Scientists from the European Nuclear Energy Agency countries will work on this project for five years. (See also Geneva Conference paper No. 314).

8 - 23 AN EVALUATION OF THE CALDER HALL TYPE OF NUCLEAR POWER PLANT
Banks, W. F., Schneider, G. A., Morgan, W. T. and Ash, E. B.
(NAA-SR-1833) Atomics International Div., North American Aviation,
Inc., Canoga Park, Calif.
January 18, 1957. 132p. Contract AT-11-1-GEN-8.

The preliminary design of a natural U, graphite-moderated, CO₂-cooled reactor and power plant similar to, but larger than, the British Calder Hall plant, with a net electrical output of 130 MWE is presented. The design is conventional, consisting mostly of standard components, the only major uncertainty being the performance of the fuel cladding. A construction cost of \$67 million and a power cost of 17 mills/kwh are estimated for this plant, assuming private utility operation in the United States, which is about three times the cost of power from conventional plants. Power output and cost for various conditions of temperature and pressure and for increased reactor size are calculated. It is estimated that a similar plant of 740 MWE output would produce power at a cost of 10 mills/kwh. Use of He as coolant is investigated and found to offer no advantage over CO₂. (auth)

8 - 24 FEASIBILITY STUDY, 44,000 KW PROTOTYPE PARTIALLY ENRICHED URANIUM,
GAS COOLED, GRAPHITE MODERATED NUCLEAR POWER PLANT (PROTOTYPE FOR
AN OPTIMUM POWER PLANT). (Cover carries title: GAS COOLED POWER
REACTOR, FEASIBILITY STUDY, 44,000 KW PROTOTYPE PARTIALLY ENRICHED
URANIUM NUCLEAR POWER PLANT).

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(IDO-2023) Kaiser Engineers Div., Henry J. Kaiser Co., Oakland,
Calif. and Nuclear Products-Erco Div., ACF Industries, Inc.,
Washington, D. C.
March 1958. Revision 1, April 1, 1958. 81p. Contract AT(10-1)-925.
Report No. 58-3-RE.

A discussion of the feasibility study and cost estimates for a gas cooled, partially enriched uranium, graphite moderated, 44 Mwe power plant is presented. The plant is designed as a prototype of a 220 Mwe plant and consists of four adjoining buildings which house the reactor plant, turbine generator,

warehouses, and offices. Other facilities include the induced draft cooling tower, a substation, and necessary utilities. The estimated costs of the 44 Mwe prototype are: design and construction \$960 per kw(gross electrical), and power produced 32 mills per kwhr of which fixed charges are 25 mills per kwhr. Outlet gas temperature and pressure of 1000°F and 387 psia give a net thermal efficiency of 35.3%. The degree of enrichment is selected to provide an average fuel element lifetime of 10,000 Mwd per metric ton. (W.D.M.)

8 - 25

FEASIBILITY STUDY, OPTIMUM NATURAL URANIUM, GAS COOLED, GRAPHITE MODERATED NUCLEAR POWER PLANT. (Cover carries title: GAS COOLED POWER REACTOR, FEASIBILITY STUDY, OPTIMUM NATURAL URANIUM NUCLEAR POWER PLANT.)

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(58-2-RE) IDO-2022, Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif. and Nuclear Products-Erco Div., ACF Industries, Inc., Washington, D. C.
March 1958. Revision 1, April 1, 1958. 88p. Contract AT(10-1)-925.

The feasibility, design, and cost estimates for a gas-cooled, graphite-moderated, natural uranium power plant optimized for minimum power cost are discussed. The plant consists primarily of one multi-purpose building which houses the reactor plant, turbine generator, warehouses, and shop. Other facilities include an office building, river pump house, substation, and necessary utilities. The estimated design and construction cost of the 220 Mwe (net) is 540 per kw (gross electrical). This excludes costs of the switchyard and transmission lines. The estimated cost of power produced is 19 mills per kw-hr of which fixed charges are 15 mills per kw-hr, based on a lifetime plant factor of 80%, annual fixed charges at 14%, and accepted utility plant accounting practices. Maximum outlet gas temperature and pressure are 800°F and 275 psia with a net plant efficiency of 31.4%. The steam power plant consists of eight steam generators which supply steam at dual pressures to a cross-compound turbine generator unit. (W.D.M.)

8 - 26

FEASIBILITY STUDY, OPTIMUM PARTIALLY ENRICHED URANIUM, GAS COOLED, GRAPHITE MODERATED. (Cover carries title: GAS COOLED POWER REACTOR, FEASIBILITY STUDY, OPTIMUM PARTIALLY ENRICHED URANIUM NUCLEAR POWER PLANT.)

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(58-4-RE) IDO-2024, Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif. and Nuclear Products-Erco Div., ACF Industries, Inc., Washington, D. C.
March 1958. Revision 1, April 1, 1958. 78p. Contract AT(10-1)-925.

The feasibility study and cost estimates for a gas cooled, partially enriched uranium, graphite-moderated, 215-Mwe power plant optimized for minimum power cost are discussed. The plant consists of one multi-purpose building which houses the reactor plant, turbine generator, warehouses, and shop. Other facilities include an office building, river pump house, substation, and necessary utilities. The estimated costs are: design and construction \$410 per kw-hr(gross electrical), and produced power 15 mills per kw-hr of which fixed charges are 12 mills per kw-hr. The 2.5% enrichment is selected to provide an average fuel element lifetime of 10,000 Mwd per metric ton. The outlet gas temperature and pressure of 1000°F and 370 psia give a net thermal efficiency of 35.8%. (W.D.M.)

- 8 - 27 FLUIDIZED BED REACTOR STUDY. REACTOR DESIGN AND FEASIBILITY STUDY
Teeter, C. L., Ciarlariello, T., Hoffman, B. L., Jorgensen, D. H.,
Judge, F. D., King, L. J., McCune, D. A., Scheve, N. R. and
Zellnik, H. E.
(CF-57-8-14) Oak Ridge School of Reactor Technology, Tenn.
August 1957. 203 p.

A feasibility study of a fluidized bed reactor system was undertaken. Initial studies were conducted on a fast, sodium cooled system, a gas cooled system, and organic and water moderated thermal systems. The organic moderated and light water moderated systems showed promise. The light water system was chosen for a detailed study to enable comparison with existing pressurized water reactors. The detailed study resulted in the selection of a slightly enriched uranium reactor with a heat production capability of 465 Mw. If the plutonium produced is recycled, natural or depleted material can be utilized as feed. The economic study indicated fuel costs between 1.8 and 3.7 mills/kw-hr assuming a plant load factor of 0.85.
(auth)

- 8 - 28 A FURTHER EVALUATION OF THE CALDER HALL TYPE OF NUCLEAR POWER PLANT
Banks, W. F.
(NAA-SR-1955) Atomics International Div., North American Aviation,
Inc., Canoga Park, Calif.
June 28, 1957. 52p. Contract AT(11-1)-GEN-8.

This study is a continuation of the work reported in
NAA-SR-1833.

The results of plant optimization studies and cost estimates of the reference design for a natural U, graphite moderated, gas-cooled reactor and power plant which was described in NAA-SR-1833 are presented. Additional engineering drawings are included, some of which were required because of changes

in turbine plant and others for estimating purposes. A construction cost estimate of \$70 million for a 165 eMw plant is presented in detail, leading to power costs of approximately 14 mils/kwh, assuming private utility operation in the United States. (auth)

8 - 29 GCR2. THE ORNL DESIGN STUDY

- - -
Nuclear Eng. 3, 277-81(1958) July.

The main features of the enriched-uranium gas-cooled graphite-moderated reactor designed by ORNL are discussed. Some of the distinctive features are enriched uranium oxide fuel, helium cooling, stainless steel clad fuel elements, and a burnup of 7000 Mwd/t. The reactor is rated at 687 Mwh and 225 Mwe net. (W.D.M.)

8 - 30 GAS-COOLED GRAPHITE-MODERATED POWER REACTORS

- - -
Power Reactor Technology, Vol. 1, No. 4, p. 53-8, September 1958.

Tabulated summaries of U. S. and United Kingdom gas-cooled power reactor design characteristics, U. S. cost estimates, and preferred fuel assemblies.

8 - 31 A GAS-COOLED LIQUID-METAL REACTOR

Falkenberry, H. L., Raseman, C. J., Robba, W. A., Sheehan, T. V. and Stoughton, L. D., Brookhaven National Lab., Upton, N. Y.
In Nuclear Science and Technology. (TID-2506), 1958. p. 125-41.

A helium-cooled liquid-metal-fuel reactor with a graphite core structure is described. The fuel is a solution of uranium in molten bismuth. The graphite is composed of blocks containing vertical holes for the fuel and horizontal holes for the helium coolant, which passes directly through the core to the turbine. The light-metal fuel circulates slowly at a rate sufficient for fuel processing. The reflector material is graphite, and, if breeding is desired, the fertile material is a thorium-bismuth slurry. A major advantage of this system lies in the ability of the closed-cycle gas turbine to efficiently utilize the clean inert high-temperature gas from a nuclear reactor in the direct production of useful power. The power-plant equipment can be made compact by the use of elevated pressures. As suitable materials of construction for higher temperatures are developed, further increases in efficiency can be realized by the closed-cycle plant. The system studies will produce 60

mega-watts at the generator terminals at an over-all efficiency of 40.4 per cent. The turbine inlet temperature is 1400°F, and the maximum system pressure is 1000 psi. The reactor core is a 5-ft cube contained in a 14-ft-diameter spherical vessel, and the maximum fuel temperature is 1800°F. (auth)

8 - 32

GAS-COOLED MARINE NUCLEAR POWER PLANT

Thompson, W. I., Adler, F., Case, K., Bond, W. H., Creutz, E. C., Hale, E. A., McClure, A. C., Nicoll, D., Nordheim, L. W., Putnam, D. F., Rostoker, N., Schnurer, G. T., Simnad, M. T., Sleeper, H. P. and Webb, G. B.
(GA-56) (Pt.1) General Atomic Div., General Dynamics Corp., San Diego, Calif.

January 25, 1957. 173p. Project 16. Contract AT(04-3)-118.

Work on the gas-cooled reactor power plant was concentrated on the following items: exploration of neutron flux and fission density in reactor core by multigroup methods, justification of 2-group procedures and selection of constants, comparison of gases for use in reactor cooling and/or power cycle, and preliminary reactor mechanical design. Detailed study was limited to the reactor type described in our proposal to the Maritime Administration, which consists of a gas-cooled, ZrH_{17} moderated core surrounded by a graphite reflector. The fuel elements are constructed of a mixture of UO_2 in stainless steel and are clad in stainless steel. In addition to the above items, a start was made on burnout calculations, control theory, prompt and delayed temperature coefficients, and reactor kinetics. The latter is particularly important in studying problems of stability and safety. The possible use of graphite instead of ZrH_{17} as a moderator was considered. Some tentative conclusions are drawn: The reactor and associated power equipment will compare favorably on a weight-per-horsepower basis with a pressurized-water type of system such as the Submarine Thermal Reactor (STR). The thermal efficiency will be on the order of 34%. The principal safety problem, for which we are now searching for a solution, is the removal of fission-product afterheat in a complete disaster. A liquid-cooled reactor has a distinct advantage because of the possibility of natural-convection cooling. The economical fuel inventory for this system is probably greater than 100 kg of U^{235} . Total costs will probably run somewhat lower than costs for the pressurized-water systems since a larger proportion of the closed-cycle plant's weight is in the shielding material, which is relatively inexpensive. (auth)

8 - 33

GAS COOLED, NATURAL URANIUM, D_2O MODERATED POWER REACTOR: REACTOR DESIGN AND FEASIBILITY PROBLEM

Dahlberg, R. C., Beasley, E. G., DeBoer, T. K., Evans, T. C., Molino, D. F., Rothwell, W. S. and Slivka, W. R.

(CF-56-8-207) Oak Ridge School of Reactor Technology, Tenn.
August 1956. Decl. with deletions February 8, 1957. 150p.

The attractiveness of a helium-cooled, heavy water moderated, natural uranium central station power plant has been investigated. A fuel element has been devised which allows the D_2O to be kept at a low pressure while the exit gas temperature is high. A preliminary cost analysis indicates that, using currently available materials, competitive nuclear power in foreign countries is possible. (auth)

- 8 - 34 GAS COOLED NATURAL URANIUM REACTORS FOR LARGE POWER STATIONS. THREE
DESIGNS
Liljeblad, R. and Madsen D. (Allmänna Svenska Elektriska AB,
Västerås, Sweden).
A/CONF.15/P/2419. 28p.

Two designs of heavy-water-moderated carbon-dioxide-cooled reactors and one graphite-moderated carbon-dioxide-cooled reactor are discussed. The designs are in a preliminary state and are to be regarded as feasibility studies. The three proposals that are described are: one reactor with 100-bar CO_2 as the coolant and of pressure tube design and one with 70-bar CO_2 and pressure vessel design, (both with heavy water moderator); and one reactor with 60-bar CO_2 as the coolant, graphite for the moderator, and a rock excavation for the pressure vessel. The plants are rated at 120 to 140 Mw(e), 180 Mw(e), and 275 Mw(e), respectively. (W.D.M.)

- 8 - 35 GAS-COOLED REACTOR FOR THE SOUTH OF SCOTLAND ELECTRICITY BOARD
Grant, P. J. (General Electric Co., Ltd., Erith, Kent, Eng.).
Nucleonics 16, No. 5, 108-13(1958) May.

In the Hunterston reactors, the 32,880 natural uranium fuel elements (1.15 in. diam x 24 in.) will be contained within a pile of graphite, and the heat generated will be transferred by circulating CO_2 under pressure. The elements consist of rods of natural uranium metal sealed in a magnesium alloy can that has heat transfer fins on its outer surface. The lattice properties were computed by standard methods and are based on exponential and fine-structure measurements. The core dimensions represent a compromise between the physical requirement of minimum over-all neutron leakage and the demand, from the heat-transfer aspect, that the gas channels should be as short as possible to minimize the pumping power. There are 208 control-rod channels in all, one in the center of each group of 16 fuel elements. The double shell construction of the containment shell eliminates the need for a thermal shield. Possible methods of fuel cycling are discussed. (W.D.M.)

- 8 - 36 GAS-COOLED REACTOR PLANTS FOR MARINE PROPULSION
Pigford, T. H., General Atomic Div., General Dynamics Corp.,
San Diego, Calif.
In Proceedings of the 1958 Nuclear Merchant Ship Symposium, Held In
Washington, D. C. (TID-7563), August 21, 1958. p.156-69.

The technical approach to the development of the Maritime Gas Cooled Reactor prototype is described. The objectives of the MGCR program are an unusually compact plant, low capital, fuel, and operating costs, and operational safety with a minimum requirement of nuclear containment. Component and material selections are discussed. Performance is estimated for the processes chosen for adaptation to the MGCR. (W.D.M.)

- 8 - 37 GAS-COOLED REACTORS
- - -
Power Reactor Technology, Vol. 2, No. 1, p. 26-32, December 1958.

Designs and performance characteristics of gas-cooled reactors as developed and employed in Europe are compared with recent U. S. developmental effort in this area.

- 8 - 38 GASEOUS-FUEL REACTOR
Baron, S.
Nucleonics, Vol. 16, No. 8, pp. 128, 130-3, August 1958.

A design study for a graphite-moderated, UF^6 -fueled, thermal reactor with a heat output of 35 MW. The fuel, at a pressure of 30 atm and a temperature of $900^{\circ}F$, is contained in double-walled aluminum tubes with helium coolant in the interspace. Plutonium and fission products can readily be separated from the UF^6 .

- 8 - 39 GRAPHITE MODERATED THERMAL REACTOR CALCULATIONS FOR THE CALDER HALL REACTORS
Mummery, P. W., Surett, J. J. and Tyrer, J. G. (Atomic Energy Research Establishment, Harwell, Berks, Eng.), and Cutts, B. (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.).
A/CONF.15/P/15. 30p.

The methods of calculation on which the Calder work was based together with their correlation with integral experiments are presented. The calculations are compared with the results of the commissioning experiments and explanations are suggested for the discrepancies. Attention is drawn to the difficulties of extending the system of calculation to reactor fuels containing plutonium and a provisional method for this is described.

A new proposal for a system of lattice calculations requiring a new correlation with integral experiments is outlined. (auth)

8 - 40

A HEAVY-WATER POWER REACTOR WITH GAS COOLING

Alikhanov, A. I., Vladimirovsky, W. V., Petrov, P. A. and Khristenko, P. I.

Soviet J Atomic Energy, Vol. 1, No. 1, p. 77-82, published as a supplement to J Nuclear Energy Vol. 3, No. 1-2, August 1956.

A power reactor using heavy water at 25 atmospheres as both moderator and coolant was designed. Aluminum alloys were used as constructional materials in the core, while the fuel elements (U^{235}) were canned in pure aluminum. Steam was raised in boilers heated by the heavy water, and was passed into steam turbines. However, the design was unsatisfactory because of the low thermodynamic efficiency of water cooling and the large amount of heavy water used.

Therefore gas was used as a coolant. The quantity of heavy water used is reduced and the thermodynamic efficiency is substantially increased. The steam parameters in the secondary circuit are increased, allowing the use of turbogenerators of higher power and medium pressure rather than low pressure steam equipment.

The coolant channels are immersed in heavy water and consequently are sufficiently well cooled on the outer surface that they can be made of aluminum alloys (the temperature of the gas leaving the channels reaches 500°C). The fuel elements, consisting of long thin rods or plates of natural uranium, are covered and sealed in a cladding of light metal which withstands a maximum temperature of 550°C . Carbon dioxide gas under pressure of 40 to 60 atmospheres was chosen as the coolant. It is proposed to commission in the Soviet Union an atomic power station of this kind with an electrical output of 100-200 megawatts. Such power stations may be expected to provide electricity at the same price as stations burning coal. (W.H.R.)

8 - 41

HIGH TEMPERATURE CERAMIC POWER REACTORS USING SiC-Si FUEL ELEMENT

Johnson, J. R., Hamner, R. L., Taylor, A. J. and Warde, J. M. (CF-56-1-30) Oak Ridge National Lab., Tenn.

January 10, 1956. Decl. with deletions January 24, 1957. 39p. Contract (W-7405-eng-26).

The high-temperature properties of ceramic materials for possible use in a gas-cooled power breeder reactor are presented. A detailed design of a gas-cooled power breeder reactor is presented. (W.L.H.)

- 8 - 42 HIGH-TEMPERATURE CERAMIC POWER REACTORS USING SiC-Si FUEL ELEMENTS
Johnson, J. R., Hamner, R. L., Taylor, A. J. and Warde, J. M.
Oak Ridge National Lab., Tenn.
In Nuclear Science and Technology, (TID-2022) (Del.), 1958.
p. 111-21; 125-31.

The physical properties of some ceramic materials with thermal neutron microscopic absorption cross sections less than 0.5 barn are surveyed for use in gas-cooled power reactors. An Si-SiC system was selected for further development. Fabrication, radiation damage, erosion-corrosion, heat flux, and penetrability studies are described. Design features of the gas-cooled ceramic power breeder reactor are described.
(T.R.H.)

- 8 - 43 HINKLEY POINT. ADVANCED DESIGN OF WORLD'S FIRST 500,000 KW NUCLEAR STATION

Atomics 8, 377-80(1957) October.

Two gas-cooled, graphite moderated reactors are proposed, similar to the Calder Hall reactors but having much greater power for the same dimensions. The gas coolant is CO₂. Six heat exchangers will be associated with each of the two reactors. A labelled cutaway drawing of one of the nuclear reactor groups is given. (M.H.R.)

- 8 - 44 HINKLEY POINT ANTICIPATED 500 MW OUTPUT

Nuclear Eng., Vol. 2, No. 19, October 1957. p. 423-6.

Features of power station for site in Somerset near Bridgewater; design provides for two graphite moderated gas cooled reactors in spherical pressure vessels and 12 heat exchangers, feeding six 93.5 Mw turbines-driving hydrogen cooled alternators generating at 13.8 kv and three 33 Mw variable speed turboalternators; fuel elements are natural uranium with rods just under 1-1/8 in. diam; pressure vessel will be 67 ft diam and 3 in. thick; other particulars.

- 8 - 45 THE HINKLEY POINT POWER STATION
Arms, H. S., Bottrell, C. and Wolff, P. H. W. (The English Electric, Babcock and Wilcox and Taylor Woodrow Atomic Power Group, Whetstone, Leicester, Eng.)
A/CONF.15/P/75. 16p.

Designs of 500 Mw nuclear power station, to be completed in 1960; design studies; site; station layout; reactor building; gas circuit; reactor core; fuelling gear; reactor control; turbine house; variable frequency sets; electrical equipment; steam dumping plant; boiler feed system; reactor shielding; schematics.

8 - 46 HINKLEY POINT, THE WORLD'S BIGGEST NUCLEAR POWER STATION

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Nuclear Power 2, 450-4, November 1957.

The layout of the Hinkley Point nuclear power station and the design of the reactors for this station are given. (W.L.H.)

8 - 47 HUNTERSTON POWER STATION

Millar, R. N. (General Electric Co., Ltd., Erith, Kent, Eng.).
A/CONF.15/P/74. 29p.

The research work necessary to enable new requirements for the Hunterston power station to be met is outlined, and a brief general description of the station is given. Components and techniques described include reactor vessel and core, fuel handling, reactor servicing machine, steam raising units, gas circuits, control and instrumentation, fuel elements, and generating plant. Operation costs are assessed, and the influence of the fuel cycle on power costs is discussed. Nuclear and design data are given. (W.D.M.)

8 - 48 INTRODUCTION TO REACTOR TECHNOLOGY

Gott, H. H.

Engineering, Vol. 180, No. 4682, p. 561-3, October 21, 1955.

Factors in designing power reactors and approach required for this work; gas cooled graphite moderated pile considered in particular. From paper before Sec G, Brit. Assn.

8 - 49 INTRODUCTION TO THE KE-ACF GAS-COOLED NUCLEAR POWER PLANT

Ritchie, J. S., Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif.

In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 1-12.

The design parameters for the various KE-ACF designs are outlined, and a summary of the design features covered in detail is presented. A description and comparison of the

preliminary design parameters and cost estimates are made for two power plants, CO₂-cooled graphite-moderated, one designed for natural uranium and the other designed for approximately 2% enriched uranium. Enrichment permits use of higher-temperature fuel element materials and also allows a smaller core volume. The resulting higher reactor outlet temperature (1000 F compared to 800 F), reactor inlet pressure (400 compared to 285 psia), and average specific power (10.3 compared to 2.4 mw/M.T.), along with higher fuel burn-up (10,000 compared to 3,000 mwd/M.T.), account for the lower power cost of 13.6 mills/kwhr for the enriched plant compared to 15.7 for the natural plant. A general description of the projected prototype helium cooled GCPR is given. (auth)

8 - 50

THE METHOD FOR THE OPTIMIZATION OF A HELIUM-COOLED GRAPHITE MODERATED REACTOR POWER PLANT

Kintner, L. L., Wood, P. M., Jackman, J. T. and Ernst, J. P. Nuclear Products-Erco Div., ACF Industries, Inc., Washington, D. C. and Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif. In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 13-48.

The optimum combination of the major design parameters was determined for the large scale helium cooled reactor power plant presently being designed by ACF Industries, Inc. and Kaiser Engineers. The basic parameters of the optimum plant including the outlet gas temperature and maximum specific power become the design basis for the Helium Cooled Prototype GCPR. The purpose of the prototype is to demonstrate the parameters of the optimum plant and to provide engineering, construction, and operating experience applicable to the optimum plant. The method and results of the optimization analysis are given. (auth)

8 - 51

METHOD OF CALCULATING DEFLECTION OF GRAPHITE SUPPORT GRID FOR GAS COOLED REACTOR

Edwards, J. C. M., Gill, S. S., Perkins, J. Civ. Eng. (Lond), Vol. 53, No. 630, p. 1400-2, December 1958.

Support grid design investigated consists of two sets of equally spaced cross beams at right angles to one another with circular ring girder around periphery; expression derived for central deflection of grid and maximum bending moment on central cross beam for range of grid diameters; comparison with flat plate theory.

- 8 - 52 NATURAL URANIUM AND HEAVY WATER REACTORS
Klik, F. and Markvart, J. (Czechoslovakia).
A/CONF.15/P/2094. 25p.

The advantages and disadvantages of using heavy water in power reactors are discussed. A detailed study of gas-and sodium-cooled heterogeneous heavy water-moderated reactors is presented. This study is concerned with the attainable reactor unit output and with the specific inventory of heavy water. The reactors are described, and the characteristics, steam cycles, fuel, etc., are discussed. (W.D.M.)

- 8 - 53 NATURAL URANIUM REACTORS FOR SMALL AND MEDIUM POWER OUTPUTS
Dayal, M., Paranjpe, S. R. and Srinivasan, M. R. (Atomic Energy Establishment, Trombay, India).
A/CONF. 15/P/1641. 15p.

It is felt that graphite-moderated gas-cooled reactors using natural uranium start becoming unattractive for power outputs below about 60 Mw(e). Preliminary design studies were conducted on BeO-moderated gas-cooled and D₂O-moderated gas-cooled reactors. The considerations affecting the design of reactors of these sizes and the effect of local conditions are set out. Moderate pressure and temperature conditions were assumed. Results of calculations on BeO-moderated gas-cooled and D₂O-moderated gas-cooled reactors for different station sizes in the range of 10 to 50 Mw(e) are presented. Moderator and fuel inventories required are given. The problems, advantages, and disadvantages of these reactor types are discussed. The economic aspects of these reactors and of conventional stations of similar sizes are examined. These comparisons should be of interest for more detailed studies. (auth)

- 8 - 54 NUCLEAR PHYSICS AT BRADWELL. SOME DESIGN ASPECTS
Lewis, S. E. (Nuclear Power Plant Co., Ltd.).
Nuclear Power 2, 414-15(1957) October.

Some design aspects of the nuclear physics at Bradwell are summarized, among these are: economic utilization studies, increasing mean irradiation, reactivity losses, and flux flattening. (M.H.R.)

NUCLEAR STUDY OF A LOW POWER BERYLLIUM OXIDE NATURAL URANIUM POWER REACTOR

Jain, R. D. and Srinivasan, K. R. (Atomic Energy Establishment, Trombay, India).

A/CONF.15/P/1945. 12p.

Nuclear study for a small-sized BeO-moderated natural uranium, CO₂-cooled power reactor was made. The core parameters, including the bucklings with and without allowances for poisoning and temperature effects, were calculated for several sets of solid rod radii and the channel and cell sizes. Fixing the core size about 3 m high and 3.6 m in diameter, with 90 cm thick graphite reflector and the rod and channel radii at 1.35 cm and 4.0 cm respectively, so as to conform with 2.54 cm thick steel pressure vessel of diameter about 6 m and the heat transfer calculations, it is found that the thermal power output, which is about 33 Mw for the uniform core, can be increased by about 20% on breaking the core into two zones, while the fuel inventory is decreased by about 9%. The extra reactivity required for poison and temperature effects is found to be about 4%. (auth)

THE ORNL GAS-COOLED REACTOR. PART 1, SUMMARY REPORT. PARTS 2, 3, AND 4. PLANT DESIGN

(ORNL-2500). Oak Ridge National Lab., Tenn.

April 1, 1958. (Pt. 1, 19p.; Pt. 2, 252p.; Pt. 3, 173p.; and Pt. 4, 185p.). Contract W-7405-eng-26.

These four parts were issued separately, but are cataloged as a unit.

A conceptual design is given for a helium cooled, graphite moderated reactor using enriched UO₂ fuel in stainless steel capsules. This study is based on existing technology. The reactor is designed to produce a gross thermal power output of 687 Mw. The net electrical power output of the power plant is 225 Mw. It is suggested that the minimum design specification for such a plant should be 1000°F gas temperature. Part II contains detailed information on gas-cooled reactor physics, materials, fuel elements, and the reactor. Part III deals with the details of the gas system, instrumentation and controls, operating procedures, and the steam system. Part IV gives the proposed plant design, the cost analysis, heat transfer performance, the reactor support and charging machine, fuel element cost data, and design data for British and U. S. gas-cooled reactors. (M.H.R.)

8 - 57

THE ORNL GAS-COOLED REACTOR. MATERIALS AND HAZARDS

(ORNL-2505) (Rev.) Oak Ridge National Lab., Tenn.
May 22, 1958. 163p. Contract W-7405-eng-26.

This report may be considered as Part 5 of ORNL-2500.

The experimental program proposed for advancing the technology of bulk oxide fuel, coolant, and the various structural materials to a stage sufficient for use in such a reactor is outlined and explained. Component tests of heat transfer systems will be operated to give a better understanding of the aerodynamics and heat transfer characteristics of the reactor system. The proper gas-handling techniques for the reactor system will be developed as an outgrowth of both in- and out-of-pile experiments with large-scale loops. The single largest developmental undertaking is the correlation of the manufacturing method for the UO_2 with its radiation damage characteristics. Statistical radiation damage testing will be conducted, with the variables being manufacturing methods of the oxide, density of the body, temperature, burn-up and stress on the element. Hazards covered include: gas stream contamination, the release of fission gases in the fuel capsule, shut-down activity and dose calculations, the removal of afterheat, nuclear excursion, the containment of radioactivity, the release of radioactivity, and the potential energy of the graphite. (M.H.R.)

8 - 58

THE ORNL GAS-COOLED REACTOR ADVANCED CONCEPTS

(ORNL-2510) Oak Ridge National Lab., Tenn.
October 2, 1958. 89p. Contract W-7405-eng-26.

A few of the uses for which gas cooled reactors may be designed are discussed and a brief commentary on several of the systems that have been suggested for these applications is presented. Fuel element design, power recovery, and instrumentation and control are discussed. To assist future design studies, the more likely possibilities for the coolant, moderator, fuel, and cladding materials are discussed with emphasis on their advantages and disadvantages. (W.D.M.)

8 - 59

THE ORNL GAS-COOLED REACTOR DESIGN

Fraas, A. P. Oak Ridge National Lab., Tenn.
In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p.119-37.

The major features of the ORNL GCR-2, a helium-cooled enriched reactor, are presented. The details of the design and the bases for the major design decisions are contained in ORNL-2500.

The major points of similarity and difference between the British designs and the GCR-2 are discussed. It appears that through use of fuel enrichment and helium-cooling it is possible to obtain both lower capital costs and lower power costs. (W.D.M.)

8 - 60

PARAMETRIC STUDIES OF THE GAS-COOLED REACTOR CONCEPT

Heacock, H. W. General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 234-42.

Generalized parametric studies of the gas-cooled graphite-moderated power reactor concept were performed at Hanford. These studies included consideration of capital charges, fuel cost, and operating expense for a number of process variables. Among the items considered for their effect on unit electrical power cost were the following: coolant gas selection, fuel enrichment, reactor thermal rating, power recovery cycle, coolant outlet temperature and reactor differential temperature, lattice geometry, and specific tube power. Subsequent to the completion of the initial studies, additional parametric studies of the reactor core configuration were performed. In these analyses attention was given to the effects upon relative unit power cost, for the more optimum cases, of the following variables; moderator to fuel ratio active zone length to core diameter ratio, specific tube power, process channel flow area ratio, and reactor coolant differential temperature. (auth)

8 - 61

A PARAMETRIC STUDY OF THE GAS-COOLED REACTOR CONCEPT

(HW-54727) (Rev.) General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

March 1, 1958. 4lp. Contract W-31-109-Eng-52.

A summary of the parametric study results, the selection of the reactor concept for further analysis in Phase III of the HAPO study program, a discussion of the parametric study, and a review of areas of process technology requiring additional study and development are presented. Rough cost estimates of 8 mils/kwh when using private financing rates of 15% capital charges per year and a by-product plutonium value of \$12/g resulted in a number of cases considered. (W.D.M.)

8 - 62

THE PLACE OF THE SODIUM GRAPHITE REACTOR IN THE U.K. NUCLEAR POWER PROGRAMME

Hill, J. M. and Fenning, F. W.

(NAPLES-UK-6) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.

(1956?). 4p.

The relative advantages and disadvantages of the sodium graphite reactor compared with gas-cooled systems in reactors for producing electricity reasonably are listed and evaluated. (F.S.)

8 - 63

POWER REACTOR STUDIES QUARTERLY PROGRESS REPORT (FOR) FEBRUARY, MARCH, AND APRIL 1957

(DP-232) Du Pont de Nemours (E.I.) & Co. Savannah River Lab., Augusta, Ga.

July 1957. 33p. Contract AT(07-2)-1.

Design concepts are given for a power reactor moderated with D_2O and fueled with natural U that will produce 100 Mw of electrical power. The basic physics and engineering problems for such a reactor are discussed, and estimates of fuel costs are given. (auth)

8 - 64

POWER REACTOR STUDIES QUARTERLY PROGRESS REPORT (FOR) MAY, JUNE, AND JULY 1957.

(DP-245) Du Pont de Nemours (E.I.) & Co. Savannah River Lab., Augusta, Ga.

September 1957. 66p. Contract AT(07-2)-1.

The physics, engineering, and equipment characteristics of six schemes for producing electric power from an atomic reactor fueled with natural U and moderated with D_2O are tabulated in detail. The tabulations were prepared in order to facilitate evaluation of various power reactor concepts and to provide sufficient data for preliminary mechanical design and cost estimation. The basis for the conceptual designs is given. Of the six power reactor concepts summarized, three are D_2O -cooled pressure-vessel reactors, one is a D_2O -cooled pressure-tube reactor, one is a H_2 -cooled pressure-tube reactor, and one is a He-cooled pressure-vessel reactor. A seventh reactor concept, a preliminary design of a CO_2 -cooled pressure-tube reactor, is discussed along with the other six. The data are somewhat preliminary in nature, and probability do not represent the economic optimum design. The experimental study of the nuclear properties of different arrangements of natural U in D_2O was extended to include oxide-fueled lattices. Preliminary measurements of the bucklings of the UO_2 -fueled lattices are summarized. Results

show that oxide-fueled lattices may be designed for long fuel-exposure cycles; however, direct substitution of oxide fuel for metal fuel in the same reactor lattice may present difficulties. Control systems are described for a D₂O-cooled pressure-vessel reactor that operates with hot moderator. A test to assess the effects of high temperature on a Zr-clad tube of natural U metal irradiated to a maximum exposure of greater than 4500 Mwd/t showed that heating to 400 to 450°C for a short time caused an additional average growth of about 0.007 inch in outer diameter. The growth that was due to irradiation at low temperature had previously been determined as approximately 0.007 inch also. The average total diametral growth due to both heating and irradiation amounted to approximately 1/2%. (For preceding period see DP-232.) (auth)

8 - 65

PRELIMINARY DESIGN, 55,000 KW PROTOTYPE NATURAL URANIUM, GAS COOLED, GRAPHITE MODERATED NUCLEAR POWER PLANT (PROTOTYPE FOR AN OPTIMUM PLANT). (Cover carries title: GAS COOLED POWER REACTOR, PRELIMINARY DESIGN, 55,000 KW PROTOTYPE NATURAL URANIUM NUCLEAR POWER PLANT.)

(58-1-RE) IDO-2021, Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif. and Nuclear Products-Erco Div., ACF Industries, Inc., Washington, D. C.
March 1958. Revision 1, April 1, 1958. 126p. Contract AT(10-1)-925.

The preliminary design parameters, cost estimates, and schedules for a gas-cooled, natural uranium, graphite-moderated power plant are discussed. The plant is designed as a 55-Mw prototype of an optimum plant rated at 220Mw and consists primarily of one multipurpose building which houses the reactor plant, turbine generator, warehouses, and offices. Other facilities considered include the induced draft cooling tower, a substation, and necessary utilities. Estimated costs are: design and construction \$990 per kw (gross electrical) and power produced 32 mills per kw-hr of which fixed charges are 26 mills per kw-hr. The physical size of the prototype is the minimum which provides sufficient reactivity to permit 3,000 Mwd per metric ton average fuel lifetime with natural uranium. The maximum operating pressure and temperature for this fuel lifetime are 285 psia and 800°F. (W.D.M.)

8 - 66

PRELIMINARY DESIGN OF A REACTOR PRODUCING PROCESS STEAM AND COBALT 60
Pierce, C. W. and Richman, D. M.
(BNL-3600) Brookhaven National Lab., Upton, N. Y.
September 1957. 20p.

The feasibility of a He-cooled, graphite-moderated reactor for use as a heat source for a gasoline fractionating unit and for production of Co⁶⁰ is discussed. The over-all

dimensions of the reactor are a core radius of 9.3 ft and a core length of 20.25 ft which must be contained in a pressure vessel designed for 10 atmosphere pressure and a temperature of about 600°F. At the design power level of 50 Mw, 1.96 megacuries of Co⁶⁰ with 15.1 c/g specific activity can be produced. An enriched 14% U-Al alloy clad with Alcoa alloy X-2219 was selected as the fuel element. The cost factors of such a reactor are detailed. (M.H.R.)

8 - 67

PRELIMINARY DESIGN PROPOSAL--DANIELS EXPERIMENTAL POWER PILE.

McCullough, C. R.

(MonN-188) (Bks. I and II) Clinton Labs., Oak Ridge, Tenn.

November 1, 1946. Decl. January 10, 1956. 178p. Contract W-35-058-eng-71.

The reactor will be a high temperature power unit designed and built especially to study experimentally the engineering features of a nuclear power plant. The pile will be run initially at 4000 KW of heat and will be designed to operate at power levels up to a maximum of 40,000 KW of heat if operating experience proves this possible. Helium gas as a coolant, will flow through the annular spaces between fuel rods and moderator bricks, and will be distributed by orifices among channels according to heat production. It will be circulated through the pile and through two external steamboilers which have a total steam capacity of 110,000 lb/hr. by 4 blowers requiring 150 KW driving power each. For the maximum anticipated heat output the pressure of the He will be 10 atmospheres and the maximum mass flow rate 34 lb/sec. Equipment will be in duplicate so that the system may be run at half power in case of partial failure. Coolant will enter the reactor at 500°F and leave at 1400°F. The highest temperature within the reactor will be 1750°F and the average reactor temperature will be 1200°F (650°C). (auth)

8 - 68

PRELIMINARY DESIGN 30,000 KW PROTOTYPE PARTIALLY ENRICHED URANIUM, Gas Cooled, Graphite Moderated Nuclear Power Plant for United States Atomic Energy Commission, Idaho Operations Office.

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(IDO-24027) Kaiser Engineers, Oakland, Calif. and Nuclear Products-Erco, Washington, D. C.

March 1959. 2 Vols. Report No. 59-3-RE.

The body of this report summarizes the recommended design for the prototype plant of the Kaiser reactor, a partially enriched uranium fueled, helium gas-cooled, graphite moderated reactor. Architectural, nuclear and thermodynamic aspects are discussed. Fuel fabrication and processing procedures are recommended. Design of utility systems, reactor safety, operation and maintenance, time and cost, and finally the performance limits of

the reactor and the necessary development program considered necessary to support the detailed design of the plant were discussed. The Appendix volume contains supporting material: outline specifications; an outline of the start-up, operation, and maintenance manual; cost estimates; construction schedules; a reactor safety study; summaries of various studies; and a set of engineering drawings.

8 - 69

A PRELIMINARY STUDY OF THE FEASIBILITY OF A GAS-COOLED, GRAPHITE-MODERATED REACTOR FOR POWER PRODUCTION. Reactor Design and Feasibility Problem.

Willmer, D. B., Crowley, J. H., Levine, Z., Mallon, R. G., Potter, M., Stichka, J. B. and Thomas, J. T.
(CF-52-8-226) (Del.) Oak Ridge School of Reactor Technology, Tenn.
August 20, 1952. Decl. with deletions February 20, 1957. 121p.

This report covers the preliminary design of a helium-cooled, graphite-moderated nuclear reactor and the associated power producing system. The problem is approached with the viewpoint that the system need not represent the economically optimum design, but one which would serve as a pilot plant for future designs. The reactor is designed to operate well within the known limits of safety for the materials of construction, using only those materials which are available at the present time. The fuel is enriched (93.4%) U^{235} in the form of UO_2 inclusions in a stainless steel matrix and with a stainless steel cladding. The reactor is rated at 200 megawatts and the net electrical power output is 55 megawatts. The helium coolant for the reactor transfers its heat to a conventional steam power cycle. The capital cost, including chemical reprocessing and fuel fabrication plants, is estimated to be \$39,000,000 and the cost of the electrical energy is 28 mils per kilowatt-hour. (auth)

8 - 70

PRESCOPING OPTIMIZATION FOR SELECTION OF ENRICHED URANIUM GAS-COOLED POWER REACTOR DESIGN

Farmers, W., Hill, E., Land, P., Trickett, K. and Matovich, S.
(AECU-3993) Nuclear Products - Erco Div., ACF Industries, Inc., Washington, D. C.
February 14, 1958. 6p. Contract AT(10-1)-925. (ACF-GCPR-101).

A preliminary set of performance calculations was made for enriched reactor cores based on five cylindrical vessel designs ranging from 13 to 36 ft in diameter. Both 7 and 19 rod UO_2 stainless steel clad duster fuel elements were employed in this survey. These two fuel element designs appeared adequate to cover the potential range of interest. A lattice spacing of 7.75 inches was employed throughout as probably leading to a reasonable uranium enrichment. The number of channels per core was established in preliminary core layouts

by the mechanical group. The inlet temperature was established 450°F, and an outlet temperature was determined for a 1300°F limiting surface temperature on the stainless steel. An optimum flow rate was established on the basis of two alternate criteria: pumping power equal to 15 per cent of gross electric power and pumping power equal to 10 per cent of gross electric power. The results of this comparison are given. In order to further select a design, capital costs per kilowatt of net electric plant power output were estimated. The performance results for the 7 rod cluster fuel element and pumping power equal to 10 per cent of gross electric power were used to make a capital cost versus vessel size comparison. (auth)

8 - 71 RELATED ARTICLES ON OAK RIDGE NATIONAL LABORATORY ORNL DESIGN STUDY

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Nuclear Eng. Vol. 3 No. 28, July 1958. p.277-85.

Details on planned GCR2 reactor using uranium oxide enriched fuel, helium cooling, stainless steel canning and having burn-up of more than 7000 MWD/t; discussion of reasons leading to concept differing from British and French projects from engineering, nuclear and economic viewpoints; GCR2--ORNL Design Study, 277-82; Design Parameters in ORNL Gas-cooled Reactor Study, 283-5.

8 - 72 RELEASE OF ACTIVITY FROM VARIOUS GCR SYSTEMS

Cottrell, W. B. Oak Ridge National Lab., Tenn.
In Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 347-62.

There are three different parameters which, as their values are increased in any particular design, tend to make the reactor more hazardous and may make containment necessary. These are operating temperature, power density, and activity in coolant. Some pertinent information relative to hazards considerations for several operating or proposed gas-cooled reactors is listed. Employing a quasitheoretical approach backed by some experimental data, an estimate of the equilibrium amount of various fission product isotopes present in the coolant was made for several fuel element models. (W.D.M.)

8 - 73 A RE-OPTIMIZATION OF SEVERAL PROCESS PARAMETERS FOR THE GAS COOLED REACTOR CONCEPT

Walker, R. C., Wade, G. E., Rudock, E. R., Massena, W. A. and Astley, E. R.
(HW-56009) General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

June 30, 1958. 93p. Contract (W-31-109-Eng-52).

The additional or supplementary conclusions regarding the gas-cooled reactor concept which have been derived from this study are: The graphite-to-uranium ratio is relatively insensitive to variations in the other process conditions considered and for most of the more optimum cases falls in the range of 70 to 80 to one (atom-to-atom ratio). Power costs are relatively unaffected by nominal differences in reactor T with optimum power costs for the cases studied falling within the range of 500 to 525°F. The length-to-diameter ratio of the reactor core appears to be a strong function of the coolant flow area ratio with optimum values in the range of 0.6 to 1.0, the lower values being associated with the smaller flow area ratios. Specific tube power is another process variable which is strongly affected by the flow area selected and for the range of combinations investigated, falls within the range of 75 to 140 kw/ft with lower values again being related to the lower flow area ratios. Initial enrichment requirements are strongly affected by an increase in the flow area ratio although less fuel is contained in the lattice for a given N_g/N_u ratio. This paradox occurs due to the fact that optimum power density increases with increasing flow channel areas. The higher fuel specific power necessitates subdividing the total fuel volume into an increased number of smaller rods which have correspondingly greater amounts of stainless steel cladding and thus higher fuel enrichment requirements. Net power costs for many of the cases studied fell within a rather narrow range; in fact, the optimum values for the three flow area ratios fall within a 0.24 mill/kwh bracket. However, this can be expected from a study of this nature and while different bases for capital and fuel cost estimates could alter the optimum ranges somewhat, the values resulting from this study are probably not too far in error since the 0.24 mill/kwh range cited above is equivalent to approximately \$500,000/yr in operating or fuel costs or \$3,350,000 in capital investment for a 300 Mw plant and the method of cost analysis assumed. (auth)

8 - 74

SCIENTIFIC PROBLEMS IN DEVELOPMENT OF NUCLEAR POWER
Cockcroft, J.

Engineer, Vol. 202, No. 5242, p. 40-2, July 13, 1956.

Problems in reactor physics, solid state physics and chemistry, to be tackled in design of stations using "improved" Calder Hall reactors and more advanced kinds of reactors. From lecture before Inst of Physics and Brit Nuclear Energy Conference.

8 - 75

SOME DESIGN CONSIDERATIONS IN GAS COOLED REACTORS

West, H. and Shaw, A. L.

Nuclear Power 3, 598-9(1958) December.

Major differences in the design of the British Mk 1 nuclear power stations are summarized. For pressure vessels there are sound arguments in favor of both the sphere and the cylinder. (J.H.M.)

8 - 76

SUMMARY REPORT ON DESIGN AND DEVELOPMENT OF HIGH TEMPERATURE GAS-COOLED POWER PILE

McCullough, C. R.

(MonN-383) Clinton Lab., Oak Ridge, Tenn.

September 15, 1947. Decl. February 27, 1957. 313p. Contract W-35-058-eng-71.

A design for an experimental nuclear power plant utilizing a high-temperature gas-cooled power pile as the energy source is described. The plant consists of the pile, a heat exchanger or boiler, a conventional steam turbine generator, and their associated auxiliaries. Helium gas under pressure transfers heat from the pile to the boiler which generates steam for driving the generator. The plant is rated at a normal output of 12,000 kw of heat and an electrical output of 2400 kw. Provisions is made for operation up to 20,000 kw of heat (4000 kw of electrical output) in the event operation of the plant proves this possible. The pile operates on thermal energy neutrons using uranium enriched to 30% U^{235} as fuel. It is a right circular cylinder with its axis vertical and contains beryllium oxide as the moderating material and permanent skeletal structure. Removable fuel elements consisting of mixtures of enriched uranium and moderating material, which may be beryllium metal, beryllium oxide, or graphite, are located in vertical channels running parallel to the axis of the cylinder. Helium gas at a maximum pressure of 10 atmospheres flows through these channels where it is heated to a maximum temperature of 1400°F. The hot gas is then circulated through a boiler where it is cooled and generates steam, thence is returned to the pile at a temperature of 500°F. (auth)

8 - 77

SUPPLEMENTARY STUDIES ON GAS-COOLED REACTOR FOR MARINE PROPULSION

Thompson, W. I., Bjørnerud, E., Chen, L. H., Hinman, G., McClure, A. C., Nicoll, D., Poppendiek, H., Rostoker, N., Schnurer, G. and Webb, G.

(GA-179) (Pt. 1) General Atomic Div., General Dynamics Corp., San Diego, Calif.

August 14, 1957. 104p. Project No. 16, Contract AT(04-3)-118.

A feasibility of a marine-propulsion power plant, based on the use of a hydride-moderated gas-cooled reactor, is presented. The principal emphasis in these studies was directed towards hazards problems, which were approached from the point of view of reactor and power-plant transient analysis and selected reactor accidents. Further efforts were devoted to certain essential design problems, including the design of equipment to remove reactor after-heat, design of control-rod drive mechanisms, and the design of fuel elements. There is also included a summary and analysis of information on CO₂, which is needed for the design of the power plant. (W.L.H.)

8 - 78 SWEDISH POWER REACTOR PROPOSAL

Liljeblad, R.

Nucleonics, Vol. 15, No. 11, p. 165-9, November 1957.

A design study for a carbon-dioxide-cooled heavy-water moderated reactor with an output of 444 mW of heat. To avoid the necessity of a large pressure vessel the UO₂ fuel elements are individually contained in pressure tubes.

8 - 79 TEMPERATURE-ZONED REACTOR

Clark, F. H. and Klahr, C. N. (Technical Research Group, New York).
Nucleonics 16, No. 5. p. 114-17 (1958) May.

The temperature-zoned reactor system attempts to improve nuclear power economics by supplying reactor heat at two different temperatures. The reactor core consists of two zones that are neutron-coupled but isolated in temperature from each other. The high-temperature zone, which produces only a small fraction of the total heat, is used to superheat steam produced by the lower zone. Separate coolant streams are used. Two temperature-zoned reactor applications have been studied: a Calder Hall type modified to produce high-temperature heat with conventional materials while retaining natural uranium as fuel, and pressurized water types operating at either lower pressure or higher power density than conventional designs. (W.D.M.)

8 - 80 TURRET: A HIGH TEMPERATURE GAS-CYCLE REACTOR PROPOSAL

Hammond, R. P., Busey, H. M., Chapman, K. R., Durham, F. P., Rogers, J. D. and Wykoff, W. R.
(LA-2198) Los Alamos Scientific Lab., N. Mex.
January 23, 1958. 26p. Contract W-7405-eng-36.

A nitrogen-cooled graphite-moderated nuclear reactor experiment is proposed to drive a closed-cycle gas turbine power plant at 1300°F. The annular core of the reactor can be rotated inside

the reflector to permit fuel loading and discharge while operating at full power. Small cylindrical fuel elements of graphite are solution-impregnated with partially enriched uranium. The fuel is recycled by incineration of the elements, chemical separation of the fission products, and impregnation of fresh graphite in a small batch process. The unclad, uncoated fuel should permit high burn-up and simple fuel processing, but allows fission product diffusion into the gas stream. While methods are proposed for the removal of these from the gas, the long-term consequences on turbine operation are unknown. The compatibility of nitrogen gas with the fuel has been studied experimentally. The radial movement of fuel gives a reactor with a constant power profile and no excess reactivity. The temperature is regulated by the fuel charging rate. (auth)

- 8 - 81 TWO EARLY FORMS OF NUCLEAR REACTOR FOR POWER GENERATION
Hinton, C.
Engineer, Vol. 200, No. 5196, p. 290-3, August 26, 1955.

How graphite moderated, gas cooled reactor came to be chosen for first British power producing pile; design is inherently stable and does not require to be isolated in interests of safety; developed first at Windscale, and then at Calder Hall; possible lines of future progress. From paper before Int Conference on Peaceful Uses of Atomic Energy, Geneva.

- 8 - 82 U. S. DESIGNS GAS-COOLED REACTORS
- - -
Nucleonics 16, No. 8, 118-21 (1958) August.

Design studies made by Kaiser Engineers with ACF Industries, Inc., as subcontractor and by ORNL on gas-cooled reactors are discussed. A summary is presented of the characteristics of the proposed AEC reactors and the United Kingdom and French reactors. (J.S.R.)

- 8 - 83 URANIUM GRAPHITE LATTICES — THE BROOKHAVEN REACTOR
Kaplan, I. and Chernick, J.
In International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Proceedings. Vol. 5: Physics of Reactor Design.
New York, United Nations, 1956. p. 295-308.

Describes the nuclear design calculations used for the Brookhaven reactor. The results of these calculations are compared with experimental results obtained during subcritical operation of the reactor, during the initial period of critical operation, and in later measurements of certain reactor

parameters. The discussion includes the multiplication factor, the migration area, the critical buckling and critical size, the effect of the central transverse gap, the excess reactivity needed, the temperature coefficient of the reactivity, and the Xenon poisoning.

8 - 84

THE USE OF ELECTRONIC DIGITAL COMPUTERS IN A STUDY OF HIGH TEMPERATURE GAS COOLED REACTORS

Wordsworth, D. V.

J. Nucl. Energy. Part B: Reactor Technology 1, 25-34 (1959) April

A review of the scope of high speed computer methods as applied to a particular reactor design study. By reviewing a particular design study the author has been able to discuss certain aspects of the work in some degree of detail, but even so it has not been possible in the space available to do more than review the types of problems to which machine methods have been applied in the pursuance of this specific study: the derivation of equations and the evaluation of results have had to be omitted. (W.H.R.)

8 - 85

TREAT, A PULSED GRAPHITE MODERATED REACTOR FOR KINETIC EXPERIMENTS

Freund, G. A., Iskenderian, H. P. and Okrent, D. (Argonne National Lab., Lemont, Ill.)

A/CONF.15/P/1848. 33p.

TREAT or Transient Reactor Test Facility is being constructed by Argonne National Laboratory to generate safely a large, transient, thermal flux (over 10^{15} neutrons/cm²) of short duration. This flux is an order of magnitude greater than that produced in the final transient of BORAX-I and will be sufficient to permit meltdown studies on mockups of fast reactor fuel elements. The practical engineering and physics design of TREAT is discussed. Attention is given the choice of cladding material, the form of the fuel element, the size of the reactor core, and the type of control rod.

9. DEVELOPMENT

9 - 1

BRITISH REACTOR OF FUTURE

Shepherd, L. R., Huddle, R. A. V., Husain, L. A., Lockett, G. E., Sterry, F. and Wordsworth, D. V.

Engineering, Vol. 186, No. 4827, p. 336-9, September 12, 1958.

Possibilities of achieving high temperatures in gas cooled reactor, discussed. From paper before Second Int. Conference on Peaceful Uses of Atomic Energy.

9 - 2

THE FURTHER DEVELOPMENT OF THE UNITED KINGDOM NUCLEAR POWER PROGRAMME

Cockcroft, J. (United Kingdom Atomic Energy Authority).

J. Brit. Nuclear Energy Conf. 3, 157-65 (1958) July.

The changes in power station performance and in the nuclear program during the last 3 years are described. The increase in power output was achieved by engineering development, with a simultaneous increase in technological knowledge. A facet of reactor technology which is receiving attention is to ensure that present and future reactors can be operated safely at all times. A detailed design study of the sodium-graphite reactor and a research study of the pressurized water reactor were carried out, and it is concluded that neither of these seem competitive with gas-cooled reactors because of maintenance problems and the cost of U^{235} . The lines of future development include development of the gas-cooled graphite-moderated reactor to maximum potential and an investigation of the advantages of heavy water as a moderator. (J.R.D.)

9 - 3

THE MARINE GAS-COOLED REACTOR PROGRAM

Bernsen, S. A., Bokros, J. C., Howard, R. C., Pigford, T. H. and Wallace, W. P. General Atomic Div., General Dynamics Corp., San Diego, Calif.

In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 243-304.

The specific objective of the Marine Gas Cooled Reactor Program is to develop, design, build, and test a prototype gas-cooled reactor closed-cycle gas-turbine power plant for maritime application. The interest in the closed-cycle gas-turbine system for the MGCR is explained, and the present status of the program is described. The goal of the system is to provide gas temperatures of 1300°F and higher and fuel element life-times in excess of one year. (W.D.M.)

9 - 4

MATERIALS SELECTIONS AND DEVELOPMENT WORK

Manly, W. D. and Coobs, J. H. Oak Ridge National Lab., Tenn.
In Information Meeting On Gas Cooled Power Reactors, Oak Ridge
National Laboratory, (TID-7564), October 21-22, 1958. p. 138-64.

The selection of the fuel, fuel capsule, pressure vessel, and related structures for the Oak Ridge National Laboratory gas-cooled reactor (GCR-2) is reviewed. Recent development work in support of the design of the forty-megawatt gas-cooled prototype reactor is discussed, including the development of an improved fabrication process for uranium dioxide fuel pellets and testing of the fuel capsule material. Preliminary results from fuel irradiation tests are presented, and the variables to be studied in the Fuel Capsule Irradiation Program are discussed. (auth)

9 - 5

WHERE REACTOR DEVELOPMENT STANDS TODAY

Lane, J. A.

Nucleonics, Vol. 14, No. 8, p. 30-7, August 1956.

Activities in United States, Russia, Britain and other countries in power and research reactor field; tabulations of power reactors listing output ratings, completion dates, etc; particulars of U S power program and some of technical developments that have stimulated work; status of research reactor progress.

10. ECONOMICS

- 10 - 1 COMPARISON OF BOILING WATER REACTORS WITH OTHER TYPES OF POWER REACTORS
Bergua, H. and Palacios, L.
Energia nuclear (Madrid) 2, No. 7, 60-7 (1958) July-September.
(In Spanish)

A comparison is made of the boiling water reactors with the pressurized water reactor and the gas-cooled, graphite-moderated reactors of the Calder Hall type. The initial cost, operating characteristics, maintenance cost, and future development are considered. (J.S.R.)

- 10 - 2 COMPARISON OF CALDER HALL AND PWR REACTOR TYPES
Puishes, A., Herron, D. P., Mash, D. R. and Webster, J. W.
(AECU-3398) American Radiator and Standard Sanitary Corp. Atomic Energy Div., Redwood City, Calif.
March 1, 1957. 80p. Project Size-up. Contract AT-(04-3)-109.

A comparison from the economic and technical stand-points is made of the British Calder Hall design and the U. S. Shippingport PWR design. The basis of comparison is an advanced design which could be placed in operation in the early 1960's. When built in the U.S., it is concluded that power costs from the two reactors are almost the same. In Britain, however, the Calder Hall type reactor would have a decided economic advantage. Cost estimates were also made for constructing these two reactors in Japan. Potentials for cost reduction are also discussed. Discussions of construction costs and power plant economics in the U.S., Great Britain, and Japan, the economic evaluation of power reactor designs, and the physics of the gas-cooled reactor are included in appendices. (B.J.H.)

- 10 - 3 THE ECONOMICS OF REACTORS OF THE CALDER HALL TYPE
Wolfe, P. H. W.
Elec. J. 160, 1729 (1958) June 6.

Using a bank rate of 6% and an efficiency of 27% and assuming a 15-year amortization, 80% load factor, 3,000 Mwd/t fuel life, and £15,000/ton fuel replacement cost, a 500 Mw nuclear power station would cost 0.425d/unit in capital costs and 0.185d/unit in fuel costs, totalling 0.61d/unit (7.2 mill). To this would

have to be added operating costs, maintenance, and time charges for the fuel on the site. The bases of these figures and the relative costs of other types of reactors are discussed. (auth)

- 10 - 4 THE FUTURE DEVELOPMENT OF GAS-COOLED REACTORS
Southwood, J. R. M. (AEI-John Thompson Nuclear Energy Co., Ltd.)
New Scientist 3, No. 71, 24-7 (1958) March 27.

When "A Programme of Nuclear Power," the program for nuclear power development in Great Britain, was published, it was thought that reactors with liquid coolants would replace the gas-cooled Calder Hall type. However, it now appears that gas-cooled reactors have a much greater potentiality for development than was thought possible. To reduce the cost of the electricity produced, the new reactors have larger cores, more fuel channels, and higher working gas pressures. The refueling can be done while the reactor is at full load. A brief comparison is made of the economics of the gas-cooled and pressurized water reactors. (J.S.R.)

- 10 - 5 NUCLEAR ENERGY FOR ELECTRICITY SUPPLY OVERSEAS
Gibb, C.
Engineer, Vol. 206, No. 5345, p. 6-10, July 4, 1958.

Probable costs of large nuclear stations; it is claimed that nuclear station employing gas cooled graphite moderated reactors complies fully with essentials for electricity generation; reactor fuels and materials; choice for future reactors. From paper before Brit Elec Power Convention, Brighton.

- 10 - 6 PWR AND CALDER HALL. HOW DO THEY COMPARE?
Herron, D. P. and Puishes, A. (American-Standard Atomic Energy Div., Redwood City, Calif.).
Nucleonics 15, No. 6, 62-5 (1957) June.

A comparison is presented of the PWR and Calder Hall Reactors as to which type of reactor has most promise for economic atomic power. The major technical, political and economic factors are considered. (W.L.H.)

10 - 7

WHEN WILL ATOMIC POWER PLANTS BE ECONOMICAL? THE PROSPECTS OF THE
GAS COOLED REACTOR

Junkermann, W.

Atomwirtschaft 3, 96-8 (1958) March. (In German)

For economy of atomic power not only is the selection of reactor type of importance, but also the development costs which must be paid and the state of the industrial production in the necessary structural parts. The example of the further development of the Calder Hall reactors shows that the concentration on only one reactor type in an expanded atomic program leads to cost reduction. (tr-auth)

10 - 8

COST AS A DESIGN CRITERION

Bender, M. (Oak Ridge National Lab., Tenn.)

In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, October 21-22, 1958. (TID-7564) p. 325-33.

The entire approach to the design of nuclear control power stations is founded on the premise of low-cost power production. In this discussion three illustrations of cost considerations in design are presented. These are capacity effects on power station capital costs, geometry effects on fuel element costs, and design standardization effects on nuclear power plant costs. (W.D.M.)

11. ENGINEERING

11 - 1 ATOMIC ENERGY SYMPOSIUM—ENGINEERING ASPECTS OF NUCLEAR POWER STATIONS

- - -

Commonwealth Engr. Vol. 46, No. 2, 3, September 5, 1958. p. 56-67, October 5. p. 74-81.

Abstracts of papers at symposium on peaceful uses of atomic energy, Sydney, June 1958. Sept: Medium Output Reactors for Australia, A. L. Shaw, 56-61; Engineering Aspects of Power Stations Employing Gas-Cooled Graphite Moderated Reactors, K. J. WOOTTON, 61-4. Oct: Hinkley Point Reactors, H. S. ARMS, H. C. BOTTRELL, P. H. W. WOLFF, 74-8; Design of Gas-Cooled Graphite Moderated Reactor, R. D. VAUGHAN, 78-81.

11 - 2 HUNTERSTON NUCLEAR GENERATING STATION—DESCRIPTION OF CIVIL ENGINEERING WORK

Dowse, J. M.

Instn Mun Engrs—J, Vol. 85, No. 11, p. 396-400, November 1958.

Characteristics and geology of site of plant; two reactors 200 ft diam and 200 ft high are built on rock foundation; cylindrical concrete container encloses reactor pressure vessel; fuel is fed by remote control by means of charge machine through 101 sleeves; 640 by 125 ft turbine hall houses 6 turbine sets of 60 Mw each; 2450-ft long twin on-shore inlet culverts, with pump house and 600-ft long tunnels out to sea supply 20 million gph of cooling water.

11 - 3 BRADWELL. INDUSTRY'S CONTRIBUTION

- - -

Nuclear Power 2, 429-37 (1957) October.

The details of some sub-contracts are given, among these are the ones for reactor standpipes, thermocouples, wired television, concrete curing, alloy steel forgings, crane motors, indicating temperature controllers, exciter air filters, gas flow measurement, water strainers, Goliath crane, pipe couplings, rotary water strainers, modulating valves, storage batteries, concrete fixings, chlorination plant, water valves rotor forgings, reduction gearing, circulator water treatment, duct hangers, condenser tubing, demineralization plant, air compressors for test, shield cooling fans, motor control gear,

temporary buildings, turning rolls for boiler making, circulator motors, gas driers, bolt heating, oil seal valves, boiler circulation pumps, forged tube bends, main gas valves, instrument air compressors, hydraulic couplings, and neutron source tubes. (M.H.R.)

- 11 - 4 BRADWELL BOILERS. THE DESIGN PROBLEMS AND SOME SOLUTIONS
Carr, F. (Nuclear Power Plant Co., Ltd.)
Nuclear Power 2, 416-19 (1957) October.

The design problems and solutions involved in the Bradwell boilers are summarized in the following topics: boiler losses, the single pressure cycle, the approach temperature, design variables, the optimum matrix, research, and development of extended surfaces. Illustrations included are a vertical section of the boiler system, boiler heat transfer losses vs. gas temperatures, power loss plotted against matrix surface, tabulated heat transfer power loss in a single-pressure cycle, and a photograph of the integral-finned heat-transfer tubing. (M.H.R.)

- 11 - 5 CALDER HALL--MECHANICAL ENGINEERING ASPECTS
- - -
Atomics 7, 354-65 (1956) October.

The reactor vessel, fuel charge and discharge, faulty fuel element detection, heat exchangers, main gas circulators and main steam turbines are described. Diagrams of steam flow, and feed water system. Based on paper 239 J/16 by R. F. Bishop et al., presented at the Fifth World Power Conference, Vienna, 1956.

- 11 - 6 DESIGNING PUMPS AND BLOWERS FOR NUCLEAR POWER PLANTS
Strub, R. A.
Nuclear Power 3, 592-4, (1958) December.

Problems in designing machines for use in the circulation of reactor coolants are discussed. Information is included on: the influence of gases on the performance of radial and axial compressors, circulation pumps, the EDF-1 axial blower, and shaft seals. (Z.H.M.)

- 11 - 7 EXTENDED SURFACE BOILER TUBING. DEVELOPMENT OF INTEGRAL FINNED
STEEL PIPE FOR BRADWELL AND OTHER POWER STATIONS
Howarth, F. (Imperial Chemical Industries, Ltd.).
Nuclear Power 2, 420-1 (1957) October.

Discussion is included on the following topics: the mild steel used, stock materials for finning, the rotary cold extrusion process, fins formed in one pass, and size specifications.
(M.H.R.)

- 11 - 8 FUTURE ENGINEERING DEVELOPMENT OF THE GAS-COOLED NUCLEAR REACTOR
Kay, J. M. (Imperial Coll., London).
J. Brit. Nuclear Energy Conf. 2, 96-100 (1958) April.

A review of the basic technology of the gas-cooled graphite-moderated reactor of the Calder Hall type is presented. It is shown that the present commercial power-producing reactor designs are based on the use of a larger size core, high average heat rating in Mw per ton of fuel, longer irradiation and slightly higher temperatures. The range of development, using materials similar to those employed at Calder Hall, is discussed in some detail. Finally, the implications of changes in materials and in basic reactor technology are analysed. (auth)

- 11 - 9 NON-NUCLEAR MECHANICAL COMPONENTS
Ernst, J. P. Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif.
In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 110-18.

The salient features of the helium prototype reactor coolant and steam systems are described. Criteria for the selection of the various components are discussed, and a brief description is given of the gas purification-vessel cooling system. The basis for selection of the steam cycle is given along with pertinent information on the steam generators, turbine and auxiliaries, and plant control. (auth)

- 11 - 10 NUCLEAR MERCHANT SHIP REACTOR PROJECT QUARTERLY TECHNICAL REPORT
(FOR) OCTOBER--DECEMBER 1957.
- - -
(BAW-1042) (Rev.) Babcock and Wilcox Co. Atomic Energy Div.,
Lynchburg, Va.
81p. Contract AT(30-3)-274.

Additional minor details of the propulsion system were modified during this period. A revised arrangement of the secondary shielding now includes a forward vestibule between the containment vessel and shield for the location of external reactor system components, particularly for the replacement of ion exchangers. Minor modifications of the containment cupola to house the control rod drives were confirmed. The basic design of most reactor auxiliary systems is complete. Only the waste collection and drain system, the control panel design, the radiation monitoring system, and the refueling system require further basic design work. The control rod effectiveness of boron--stainless steel rods with stainless steel fuel element cans was evaluated. As a result, the range of consideration of burnable poisons and soluble poisons likely to be required was more firmly established, and confirmation of exact values required awaits only operation of the critical experiment. Arrangements were made to construct a small-scale model of the containment vessel and its internal components to verify erection procedures and clearances between equipment for maintenance. Further studies were made on the optimization of the primary and secondary shielding in preparation for conferences on secondary shielding design with the newly selected ship-builder. Modifications to the reactor control system were developed to provide maneuvering rates comparable with other merchant ships; the incorporation of these modifications is yet to be negotiated with the AEC. The design of the buffer seals was developed to the point where buffer charge pumps could be sized and their specifications issued. (For preceding period see BAW-1023) (auth)

11 - 11 NUCLEAR REACTOR PROGRESS REPORT

(BNL-21) Brookhaven National Lab., Upton, N. Y.
October 1, 1948. 134 p. Decl. February 7, 1957.

Progress during the preceding 9-month period on construction of the Brookhaven Graphite Reactor is reported. Engineering studies on pressure drops, heat transfer characteristics, graphite oxidation rates, slug design and fabrication, charging technique, helium systems, etc., are summarized. Calculations by the Theoretical Group on revised pile constants, startup and shutdown problems, control, and other problems are included. Meteorology studies are reported at considerable length. (G.Y.)

11 - 12 REPORT ON THE ENGINEERING TESTS MADE IN CONJUNCTION WITH THE DESIGN OF THE BROOKHAVEN NUCLEAR REACTOR
Winsche, W. E., Bareis, D. W., Karp, J. and Page, W. R.

(BNL-23) Brookhaven National Lab., Upton, N. Y.
October 1, 1949. Decl. September 21, 1956. 70p.

The experimental work done with full-scale and small-scale mock-ups of sections of the Brookhaven Reactor cooling system indicates the split flow design to be feasible. A method was devised for calculating the air velocity and pressure pattern in the reactor slot for any combination of slot width, total air flow rate, reactor power level, and loading pattern. Consequently, a method for determining the required orifice pattern for any combination of the above conditions was developed. Tests performed on the reactor fuel assembly indicate that the heat transfer and air pressure drop characteristics may be correlated by the familiar empirical relations. Test results show that the fuel assembly will be physically stable under the conditions expected in the Brookhaven Reactor. (auth)

11-13

SOME ENGINEERING PROBLEMS IN DESIGN OF GAS-COOLED GRAPHITE-MODERATED POWER REACTOR

Lindley, A. L. G.

S. African Mech Engr. Vol. 6, No. 9, p. 381-98 (discussion) 398-406.
April 1957.

Design principles for reactor pressure vessel; advantages of using double shell construction; need for steels with low creep properties and low ductile to brittle transition temperature; problems inherent in design of gas circuit involving unusual design of gas blowers and gas tight valves; design of fuel elements; system for charging and discharging fuel whilst system is on load; control of reactor.

11 - 14

SPECIAL BRADWELL ISSUE

- - -

Nuclear Power Vol. 2, No. 18, p. 401-37, 1 supp chart. October 1957.

Group of related articles covering power station at Bradwell in Essex; Civil Engineers, A. H. Brown; Bradwell: Progress Report; General Description of Station; Nuclear Physics, S. E. Lewis; Boiler Design, F. Carr; Integran Tubing, F. Howarth; Turbines and Blowers; Layout of station, T. Andrews; Boiler-making for Bradwell; Special Lubricacants; Industry's Share-Details of home Contracts; List of Contractors; large cutaway color illustration showing Bradwell internal parts.

11 - 15

SUPPORTING SKIRTS FOR REACTOR PRESSURE VESSELS

Hicks, R.

Engineer, Vol. 204, No. 5300, August 23, 1957. p. 256-8.

Problem encountered during design of vessel for nuclear power station for South of Scotland Electricity Board; how general solution was obtained to give moments and shear forces induced in vessel and its supporting skirt due to various types of loading imposed on structure during fabrication and full operating conditions.

11 - 16 VESSEL AND HEAT EXCHANGER OF THE E.D.F.1 CHINON POWER PLANT
Lamiral and Lancel (Electricite' de France, Paris).
A/CONF.15/P/1199. 28p.

The 4.2 in. thick steel pressure vessel for the gas-cooled EDF-1 is described. The dimensions of the pressure vessel (cylinder with hemispherical ends 33 ft diameter, 76 ft high) were fixed by the volume of the active core and its reflector. In view of the characteristics of the CO₂ (140°C outlet) the number of vaporization stages for the heat exchanger was fixed at two. The steam pressure is kept below that of the CO₂. The multiple exchanger consists of 120 cylindrical elements of 24 in. diameter and 9.2 ft high. (W.D.M.)

12. EXPERIMENTAL FACILITIES

- 12 - 1 A CRYOSTAT FOR REACTOR IRRADIATIONS IN LIQUID NITROGEN
Thompson, M. W. and Jefferson-Loveday, D. W.
J. Sci. Instrum., Vol. 35, No. 11, 397-9 (November 1958)

A liquid nitrogen cryostat which has operated inside the BEPO reactor at Harwell is described. Provision was made for removing irradiated specimens in liquid nitrogen after periods of up to one month. Some difficulties encountered and the methods used to overcome them are discussed.

- 12 - 2 DISPOSITIF D'IRRADIATION BIOLOGIQUE A LA PILE EL 2. (A Device for Biological Irradiation in the EL 2 Reactor at Saclay.)
Fitoussi, L. and Nichon, G.
(CEA-724) France. Commissariat a'l' Energie Atomique, Paris.
1957. 5p.

The development of a special device adapted to an experimental channel of the EL2 reaction is presented. This device allows, during normal operation of the reactor at high power, to put into the experimental channel, and to extract from it, a cradle designed to contain objects or small animals to be irradiated. The device makes possible short-time irradiations in EL2. Particular care was given to the radiation shield. (auth)

- 12 - 3 A PNEUMATIC SYSTEM FOR IRRADIATION IN A NUCLEAR REACTOR
Haeffner, E. and Jonsson, et al. (AB Atomenergi, Stockholm).
A/CONF.15/P/158. 12p.

The system used for irradiations in the Swedish reactor R1 is described. The target materials are hermetically enclosed in aluminum cans. The irradiation arrangement consists of three movable cylinders, each of which is put in a horizontal channel through the reactor and operated by a pole sticking out through the reactor wall. The cylinders each contain 20 bores in which the cans fit. A loading tube leads to the channel from above and an unloading one from below. By pulling the pole it is possible to place any bore straight in front of the loading tube and a sample can be blown in by compressed air. To remove the can, the cylinder is moved and turned 180° which causes the can to drop into the unloading tube. Samples can be inserted or removed while the reactor is operating at full power. This means that the irradiation time can be chosen

arbitrarily and that the arrangement has a great capacity for short time irradiations. Radioactive samples are not taken out into the reactor hall but are sucked by vacuum from the reactor to a bank where they are stored behind a radiation shield. Holders are fastened to an endless chain so that the samples can be taken in and out independently. From the bank the samples are transported to a hot laboratory or to a handling cupboard where they are transferred to shipping containers. The samples may also be taken directly from the reactor to the laboratory or the cupboard. (auth)

13. FUEL ELEMENTS

13 - 1

ALPHA CANNING OF URANIUM SLUGS FOR THE ORNL GRAPHITE REACTOR
Cunningham, J. E. and Adams, R. E. Oak Ridge National Lab., Tenn.
In Proceedings of the French-American Conference On Graphite Reactors,
(Held At Brookhaven National Laboratory), (BNL-489), November 12 to 15,
1957. p. 174-86.

The significant factors affecting the performance of the aluminum-clad uranium fuel elements used in the ORNL Graphite Reactor are reviewed. The important requirements of a diffusion barrier, a metallurgical bond, and a leak-proof closure were achieved for the present reactor loading by using canning techniques in which the heat-treated uranium slugs were dipped into a molten aluminum-silicon alloy bath. A diffusion-resistant layer of $U(Al, Si)_3$ compound forms at the uranium surface. The coated slug, the cap, and the can are assembled while submerged in a second aluminum-silicon alloy bath, and are brazed together with a thin layer of aluminum-silicon eutectic alloy to form a thermal-conducting bond and a tight closure. The techniques used in the canning operation are described and discussed. The performance of the fuel elements, from the aspect of resistance to failure in reactor service, is briefly reviewed. Information is presented on the radiation-induced growth which occurs. A significant cause of rupture is traced to longitudinal growth of certain lots of slugs which were not correctly beta heat-treated prior to canning; slugs correctly heat-treated show very little growth. (auth)

13 - 2

THE ASSESSMENT AND TESTING OF FUEL ELEMENTS
Hardy, H. K. and Lawton, H. (United Kingdom Atomic Energy Authority, (Springfields, Lancs, Eng.)).
A/CONF.15/P/306. 24p.

Recent U.K. Atomic Energy Authority experience in assessing the performance of fuel elements for power reactors is described, together with examples of the experimental work conducted out-of-pile. A theoretical appraisal was made of the factors which may limit the working life of the reactor under the design conditions or which may make large-scale manufacture of the fuel elements difficult or costly. The analysis pays attention to the sources and magnitudes of the strains likely to arise in the fuel elements. Irradiation effects on metallic uranium fuel include changes in macroscopic shape due to preferred orientation, volume swelling caused particularly by gaseous fission products, the development of surface irregularities associated with individual grains, and irradiation creep due to deformation while undergoing fission under applied loads.

Strains may also be caused by stresses due to temperature gradients, by thermal creep under applied load and by thermal cycling. Strains in the can arise from the accommodation of the deformation of the fuel and are created by the forces due to the coolant flow. Experiments were conducted in-pile to establish the basic phenomena both at Windscale and A.E.R.E., Harwell. Creep rates were determined by tests of samples and by tests on the actual components, both separately and after assembly. The behavior of the fuel elements during a number of the postulated hypothetical reactor faults is of interest. The behavior of the Calder Hall reactors and fuel elements were examined and the implications taken into account. The experimental work includes the proving of the production techniques for the fuel elements. Non-destructive testing methods are used to establish the soundness of the materials of the final elements. The influence of the production methods on the performance of the fuel elements has been considered in the analysis and by experimental checks. (auth)

13 - 3

AUTOMATIC WELDING OF FUEL ELEMENTS

Briola, J. (Commissariat à l'Energie Atomique, Paris).
A/CONF.15/P/1161. 33p.

The sealing of cartridges by automatic welding is described. The techniques include inert-gas welding used for G-2 reactor cartridges, inert atmosphere arc welding done in an argon-filled plastic vessel where easily oxidized metals are involved, and electronic welding, recommended for small elements used for special purposes and for zirconium. (J.R.D.)

13 - 4

DATE PROCESSING FOR BURST SLUG SYSTEMS

Bowen, D. D.
Nuclear Power 2, 91-5 (1957) March.

A system is described for detecting burst slugs in gas-cooled power reactors. (W.L.H.)

13 - 5

DEFLECTION OF FINS IN CALDER REACTORS

Hall, W. B.
(IGR-TM/W-057) Gt. Brit. Windscale Works, Sellafield, Cumb., England.
May 16, 1956. 7p. (IGC-NPCC-FEWP/P-56).

Both the heat transfer and pressure drop characteristics of the Calder reactor fuel elements have been found to be markedly affected by a slight bowing of the fins. It is shown that there will be no significant creep deflection of the fins in the Calder reactor as a result of stress imposed by the coolant flow. (M.H.R.)

13 - 6

THE DETECTION OF BURST FUEL ELEMENTS IN NUCLEAR REACTORS

Cunningham, K. W., Hough, G. V. and Aliaga-Kelly, D. (Plessey Nucleonics Ltd., Northampton, Eng.).

A/CONF.15/P/57. 25p.

A general survey is made of the current practice of burst fuel detection systems and the problems involved. A description is given of the Calder Hall equipment, followed by the improvements which are being introduced for the first stage C.E.A. nuclear power stations. The corresponding requirements for water cooled reactors are considered and some details given of apparatus which has operated successfully on a research reactor. An outline is given of some of the problems which confront future designs and some possible ways in which solutions may be obtained. (auth)

13 - 7

DEVELOPMENT OF MANUFACTURING TECHNIQUES FOR THE CALDER HALL FUEL ELEMENT

Butler, F., Harper, J., Morrison, L. H. and Pardoe, J. A. (United Kingdom Atomic Energy Authority, (Risley, Lancs, Eng.)).

A/CONF.15/P/317. 14p.

A general description of the cartridge is given with an indication of the stages leading up to its production, including feasibility studies, technical development, and a prototype production line. The cartridge is described, together with the method of manufacture and heat treatment of the uranium bar and the factors which led to the route selected. Conditions which gave rise to the design of can are traced and the methods of manufacture reviewed. In particular, stress is laid on the advance in fabrication of magnesium alloys which canning demands have promoted. Methods of locking to check ratchetting are discussed, and development work aimed at improved versions of the cartridge is described. The philosophy which led to the design of closure selected is given, as well as the reasons for choosing a particular welding technique for the final seal. Testing methods and helium filling are described. (auth)

13 - 8

THE DEVELOPMENT OF THE HUNTERSTON FUEL ELEMENT

Wootton, K. J. and Dennis, W. E. (General Electric Co., Ltd., Erith, Kent, Eng.).

A/CONF.15/P/1523. 21p.

In order to extend temperature rating and service life obtainable from a fuel element consisting of natural U canned in Mg alloy beyond those being achieved in the Calder Hall Reactors, significant changes in the design were deemed to be necessary. The reasons for these changes are discussed and the metallurgical and heat transfer investigations which have made these

developments possible are described. The evolution of the processes for reducing the permeability of the graphite support tubes is described. Experiments are described which were designed to determine the possible modes of failure of the various components and their effect of the reactor operation. (W.D.M.)

- 13 - 9 EXAMEN DES CARTOUCHES D'URANIUM SORTANT DE EL2. (Examination of Uranium Slugs Removed from EL2).
Cherel, G. and Bloch, J.
(CEA-688) France. Commissariat a l'Energie Atomique, Paris. June 15, 1957. 26p.

The uranium slugs from the various sets of rods in EL2 have been examined. This report gives the results of the first observations relating to distortions and surface condition of the slugs, and to modifications in the micro-structure of the uranium. (auth)

- 13 - 10 FABRICATION OF THE FOURTH SET OF FUEL ELEMENTS FOR THE EXPERIMENTAL PILE EL2
Ringot, C. France. Commissariat a l'Energie Atomique, Paris.
In Fuel Elements Conference, Paris, (TID-7546), November 18-23, 1957.
p. 182-227.

The fabrication is described in detail. Topics covered are: the binding problem, the connection between can and rod, the basic materials: uranium, cans, and canning operations and welding, manufacturing controls, and mounting in the reactor. The various processes and pieces of equipment are illustrated. (M.H.R.)

- 13 - 11 FUEL ELEMENT BEHAVIOUR UNDER IRRADIATION
Eldred, V. W., Greenough, G. B. and Leech, P. (United Kingdom Atomic Energy Authority, Sellafield, Cumb., Eng.).
A/CONF.15/P/50. 23p.

Changes which occur in fuel elements during irradiation in the Windscale or Calder Hall reactors are described. Particular attention is paid to behavior of the natural uranium fuel. Surface wrinkling is shown to be dependent on irradiation temperature as well as grain size. The microstructure of irradiated material is described. The slight decreases in density that occur also vary with irradiation temperature and are partly due to the development of small irregular voids in the metal. Changes in length and diameter are normally very small. Of special interest is the bowing of vertically stacked Calder Hall fuel elements. It is shown that the cartridges bow at a rate in agreement with the original predictions. Examination of aluminum cans after irradiation has revealed a slight

increase in hardness but otherwise little change. The magnesium alloy used for the Calder Hall cans does not increase in hardness under similar conditions but may develop small grain-boundary voids if slowly strained, especially at low temperatures. A brief summary is given of some of the important causes of the cartridge failures that have been examined. (auth)

- 13 - 12 FUEL ELEMENT DESIGN
Lindley, P. A.
Atomic Energy Rev. 1, 13-15 (1957) March.

The factors that influence the design of fuel elements for gas-cooled graphite-moderated thermal reactors using natural U are described. (W.L.H.)

- 13 - 13 FUEL ELEMENTS FOR A HIGH-TEMPERATURE REACTOR
Boettcher, A. and Wirths, G. (Degussa, Frankfurt a.M.); and
Schulten, R. (BBC, Mannheim, Ger.).
A/CONF.15/P/1005. 5p.

Fuel elements for a high-temperature inert-gas-cooled reactor are described. Requirements include high mechanical strength, 1100°C surface temperature, resistance to sudden thermal change, and ease of charging and discharging. The fuel elements consist of spherical graphite bodies with a diameter of 5 to 6 cm and a borehole into which the fuel is inserted. The borehole is tightly sealed off by a graphite plug. (W.D.M.)

- 13 - 14 FUEL ELEMENTS FOR ORNL GRAPHITE REACTOR
Boyle, E. J. and Cagle, C. D.
(CF-52-6-10) Oak Ridge National Lab., Tenn.
June 3, 1952. Decl. February 7, 1957. 13p. Contract (W-7405-eng-26).

Canning procedures, testing methods, and operating experience for unbonded Al-jacketed slugs and alpha canned Al-Si bonded slugs are given. The alpha-canned Al-Si bonded slug proved most satisfactory for use in the ORNL Graphite Reactor.
(D.E.B.)

- 13 - 15 THE FUEL ELEMENTS IN PRESSURIZED GAS REACTORS
Stohr, J. A., Englander, M. and Gauthron, M. (Commissariat à l'Énergie Atomique, Paris).
A/CONF.15/P/1157. 74p.

Various cladding materials and processes and natural uranium base fuels were investigated for use in CO₂ cooled reactors. Tests used in connection with the fuel element study are described. The processing of the Saclay EL-2 fuel elements are described, from the vacuum casting of the raw billet to the inspection tests prior to loading into the pile. The operational results of the first set of fuel elements that have been used in EL-2 are outlined, together with a description of the few incidents which have occurred during operation. Fuel elements for the reactors now in the building stage are described, and the change in design discussed. (W.D.M.)

- 13 - 16 GAS-COOLED REACTOR PROJECT. SEMI-ANNUAL PROGRESS REPORT FOR THE PERIOD ENDING DECEMBER 31, 1958.

(ORNL-2676) Oak Ridge National Lab., Tenn.
March 16, 1959. 165p. Contract W-7405-eng-26.

"Most of the activities described were initiated in support of the prototype gas-cooled power reactor (GCPR) designed by Kaiser Engineers - ACF Industries. Some advanced reactor design studies were made, and basic design studies required to facilitate design review work were completed that have application beyond the limited review objective.

In furtherance of an assigned responsibility for providing the fuel elements for the GCPR, methods for fabricating and testing fuel elements were developed. Tests of mechanical properties were undertaken as required to obtain the information needed in the development of the fuel elements. In-pile and out-of-pile tests of the components and materials of the reactor core were also initiated." (auth) (For preceding period see ORNL-2500)

- 13 - 17 GAS MONITORING EQUIPMENT FOR BERKELEY POWER STATION
Harlen, F. (Dowty Nucleonics, Ltd.)
Nuclear Eng. 3, 158-62 (1958) April.

The gas monitoring equipment to detect faulty fuel elements for the Berkeley power station is described and compared to the Calder Hall equipment. A discussion of future developments in gas monitoring equipment is presented. (W.D.M.)

- 13 - 18 IMPROVEMENTS IN OR RELATING TO FUEL ELEMENTS FOR NUCLEAR REACTORS
Hall, W. B. and Davis, T. (to United Kingdom Atomic Energy
Authority).
British Patent 789,257. November 5, 1955. Nuclear Power 3, 142
(1958) March.

It is a known fact that as heat ratings in gas-cooled reactors increase, the optimum fin height on the fuel elements becomes shorter and shorter. Here is a glimpse of a possible advanced type of fuel element for a gas cooled reactor of high heat-rating. The inventors say that as the need for deep finning is reduced, so it becomes possible to consider accepting fuel element sheath materials having a reduced thermal conductivity and an increased neutron absorption so long as compensating advantages such as improved mechanical properties at high temperatures are obtained. One possibility is stainless steel. One form of element designed accordingly is described. The 'fins' are frustoconical in shape which give the element a different heat transfer performance according to which way up it is installed in the fuel channel. It is proposed to make these elements by casting the bar roughly to shape and then machining it. The sheath, in cylindrical form, would be slipped over the uranium bar and then rolled on to shape. End fittings are similar to those used at Calder but could obviously be adapted to particular needs. To increase turbulence, the sheath surface can be roughened and rings can be inserted in the channel. Another type of element to achieve the same object is also described.
(auth)

- 13 - 19 IMPROVEMENTS IN OR RELATING TO NUCLEAR REACTORS
Moore, R. V. and Packman, G. (to United Kingdom Atomic Energy
Authority).
British Patent 779,408. July 17, 1957. Nuclear Power 2, 394 (1957)
September.

As is now well known, the charge in a Calder Hall fuel element channel consists of six magnesium-canned uranium bars stacked one above the other: this patent gives more details. A basic idea of the invention is the preferential cooling of parts of fuel element cans which are subject to high gravity stresses from the vertical stacking arrangement. This allows a higher temperature and greater output. Build up of heat by conduction from element to element is induced by the inclusion inside the cans of a dish of magnesium oxide. In this way the parts of the fuel cartridges subjected to high stresses are kept at relatively low temperatures. Various minor alternative arrangements are included in the specification. It should be noted that recently, owing to the so-called 'Cottrell creep,' it was found that the lower elements at Calder were tending to bow in the middle. This has been cured by fitting some kind of additional spider in the center of the affected elements. (auth)

13 - 20 NOVEL FUEL ELEMENT FEATURE OF SCOTTISH PLANT

Nuclear Power, Vol. 2, No. 10, p 42-7, February 1957.

Location of South of Scotland Electricity Board's station is planned for site on Firth of Clyde; entirely novel method, GEC Replaceable Channel Fuel Element, developed for support of 10 elements constituting complete charge for one vertical channel; inner shell; on-load fueling; frequency controlled rods; burst slugs; steam raising; gas ducting; centrifugal blowers chosen; dual pressure steam cycle; control and instrumentation.

13 - 21 PRECISION VACUUM CASTING OF PIPPA TYPE BARS

Lloyd, H. and Symonds, A.
(AERE-M/R-1608) Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.
February 7, 1955. 23p. (TRDC-P/109).

Using otherwise standard vacuum techniques, PIPPA type U bars were cast in Cr plated mild steel moulds with alumina and benzene smoke for the mould dressing. The "as cast" surface finish of the bars, particularly those from the smoke dressing, was superior to any previously obtained. In view of the diametral consistency between bars and along the length of individual bars and the high quality surface finish, machining may be unnecessary. This consideration and the extended life of these moulds by comparison with alternative materials make their use economically attractive. (auth)

13 - 22 A PRELIMINARY SURVEY OF THE ADVANTAGES OF INCREASING THE URANIUM ROD DIAMETER IN THE WINDSCALE PILES

Gallie, R. R. and Hall, W. B.
(RDB(W)/TN-57) Gt. Brit. Windscale Works, Sellafield, Cumb., England.
March 11, 1953. 6p. (WTSC/R-41).

It is shown that a gain of at least 8.5 per cent in plutonium production can be achieved by using uranium rods of 1.2 inches diameter. In addition, there would be a money saving of 80,000 pounds per annum on enriched metal. (auth)

13 - 23 PRESSURE DROPS AND LIFTING EFFECTS OF STACKED FUEL RODS IN A SIMULATED PILE FUEL CHANNEL. TESTING REPORT NO. 1

Fromm, L. W.
(M-3956) Clinton Labs., Oak Ridge, Tenn.
February 11, 1947. Decl. February 7, 1957. 12p.

Graphite rods in a simulated fuel channel were investigated to show pressure drops and lifting effects. Correlations were made to conditions in the proposed Daniels Power Reactor. (F.S.)

- 13 - 24 PROGRESS RECENTS DE LA DETECTION DES RUPTURES DE GAINES DANS LES REACTEURS FRANCAIS, G1, EL2, G3, EL3. (Recent Progress in the Detection of Bursts in the Canning in French Reactors, G1, EL2, G3, EL3.) Goupil, J., Grenon, M., Raffailhac, J. and Roquin, A. (Commissariat à l'Énergie Atomique, Paris). A/CONF.15/P/334. 46p.

In heterogeneous reactors the fuel is protected from contact with the coolant by a coating. Any fissure or break in this canning must be detected so that excess reactivity can be avoided. The methods and apparatus used in the gas-cooled reactors G-1, G-3, and EL-2 and in the water-cooled reactor EL-3 are described. The method is based on the detection of fission gases. An instrument named the evolution meter is designed to make the measurement of the activity of the coolant gas in the channels by a zero method, and it is independent of the activity of the coolant itself and of the uranium contamination of the canning. In the EL-3 the detection system is based on measurements by G-M counters of the activity of the fission gases carried by diluted helium into the heavy water and then extracted by hydrocyclones. (J.S.R.)

- 13 - 25 ROD FUEL ELEMENT FOR GAS-COOLED HEAVY-WATER POWER REACTOR Christenko, P. I. and Petrov, P. A. et al. (U.S.S.R.) A/CONF.15/P/2053. 24p.

The fuel element consists of a bunch of thin uranium pins. Each pin has a protective sheath made of Mg - Be alloy. The bunch of pins is placed in a tube made of Mg alloy. The radiation growth of U is eliminated by tempering the pin in the beta phase. Buckling of the pins is eliminated by load annealing in the vertical position. Canning of the pin ends is accomplished by welding steel tips, which simultaneously serve to mount the pins. (W.L.H.)

- 13 - 26 SAFETY OF MAGNESIUM CANNING FOR CO₂-COOLED REACTORS Salesse, M. Nucleonics 16, No. 2, 123-4 (1958) February.

The French CO₂-cooled reactor EL underwent a series of temperature oscillations. When the four fuel elements were later removed, it was found that on two of the rods the Mg canning

had partially melted. Molten Mg had flown away and later solidified around the melted area forming cushions and drops. The surface of the grooved U was undamaged. From these results it appears that commercially pure electrolytic Mg is a safe canning material in CO₂-cooled reactors. (J.S.R.)

13 - 27 SELECTION OF FUEL-CANNING MATERIALS FOR CARBON-DIOXIDE-COOLED NUCLEAR REACTORS

Woodfine, B. C. (General Electric Co.).
Atomic Energy Rev. 1, 91-6 (1957) September.

The various properties that a fuel element canning material must possess for a carbon-dioxide-cooled nuclear reactor of the Calder Hall type are discussed. (W.L.H.)

13 - 28 SPIRAL FUEL ELEMENT FOR GAS-COOLED REACTORS

Dahlberg, R. C. (Knolls Atomic Power Lab., Schenectady, N. Y.) and Evans, T. C. (U. S. Army Corps of Engineers, Washington).
Nucleonics 16, No. 4, 106-8 (1958) April.

The fabrication and utilization of spiral fuel elements for gas-cooled reactors are described. The fuel element assembly consists of two parts: the fuel spiral and the process tube. The spiral fuel element is fabricated by cladding both sides of a thin uranium sheet and brazing parallel zirconium wires to one side of the plate in the lengthwise direction. The assembly is rolled to form a spiral with the zirconium wires maintaining a constant separation of the plate surfaces. The spiral is placed in the process tube which separates the spiral and coolant from the moderator and, if necessary, provides a pressure tube for the coolant. The unique characteristics of the spiral fuel element are briefly discussed. (J.S.R.)

13 - 29 SUPPLEMENTARY COMMENTS TO ORNL CF-58-5-97

Epel, L. G.
(CF-58-8-37) Oak Ridge National Lab., Tenn.
August 12, 1958. 14p. Contract (W-7405-eng-26)

In a previous report (ORNL CF 58-5-97) the temperature structure in the GCR-2 fuel elements and coolant stream was presented. The analysis depended on several assumptions, two of which were more controversial than the others. One was the stipulation that the "equilibrium" temperature profile (i.e., a temperature profile that did not change with axial location) had been achieved at the cross section being studied. The other simplification made was that the coolant could be thought of as consisting of two regions of constant thermal conductivity, one being the

usual thermal conductivity and the other being a space average of conductivities found by using the Martinelli analogy which attempts to take into account the effects of eddy diffusivity. This report lends validity to the first assumption and verifies that the effect of the latter mentioned approximation is as was suggested in the aforementioned report. To make the problem more tractable analytically, one dimensional rectangular geometry was chosen. (auth)

13 - 30 TECHNICAL SCOPE OF GAS-COOLED REACTOR FUEL ELEMENT IRRADIATION PROGRAM

(CF-58-8-4) Oak Ridge National Lab., Tenn.
August 1, 1958. 22p. Contract (W-7405-eng-26).

A set of 55 experiments has been outlined to provide a minimum irradiation program for selection of UO_2 pellet geometry and fabrication techniques, and canning technology. These experiments fall into three categories: prototype units in which radial dimension and heat fluxes are close to proposed design values, but irradiation times are long; reduced-size prototype for accelerated tests in which most variables will be studied; and miniaturized pellet irradiation to obtain high burnup for fission gas release studies. Reactor space has been found generally available and several installations are now examining their capabilities to participate in the program. A tentative schedule has been drawn to illustrate the feasibility of the program. (auth)

13 - 31 TEMPERATURE STRUCTURE IN GAS COOLED REACTOR FUEL ELEMENTS AND COOLANT CHANNEL

Epel, L. G. and Furgerson, W. T.
(CF-58-5-97) Oak Ridge National Lab., Tenn.
May 27, 1958. 24p. Contract (W-7405-eng-26).

An analysis of the temperature structure in the GCR-2 fuel elements and coolant stream at the position where the maximum fuel element surface temperature exists is presented. Results were obtained by numerical methods on the IBM 704 digital computer. The effect of variation in channel size is shown, and a method of data correlation is suggested. Preliminary conclusions are presented as to the effect of temperature structure on the design and testing of gas cooled reactor fuel element configurations. In view of the rate at which gas cooled reactor work is proceeding results of the calculations are being published in their current incomplete form. The study is continuing in an effort to refine the calculations, and experimental data will eventually be available with which to verify the analytical conclusions. (auth)

14. FUEL SYSTEMS

- 14 - 1 ANALYSIS OF THE MAXIMUM THERMAL STRESSES GENERATED IN THE FUEL TUBES DURING STEADY STATE OPERATION OF A HIGH TEMPERATURE PILE
Sibbitt, W. L. and Etherington, H.
(MonN-292) Clinton Labs., Oak Ridge, Tenn.
December 16, 1946. Decl. February 28, 1957. 34p.

A theoretical investigation was made of the probable thermal stresses generated in cylindrical BeO and graphite fuel tubes during steady state operation in the Daniels Power Pile (Design 1). Various tube dimensions were studied over a wide range of power levels, and resultant data are given. (B.J.H.)

- 14 - 2 BURN-UP CALCULATION OF NATURAL-URANIUM GRAPHITE-MODERATED POWER REACTOR BY USING DIGITAL COMPUTER
Goto, M., Yamada, T., Takegoshi, T., Fujinaka, M. and Morita, T.
(Electrotechnical Lab., Industrial Science and Technology Agency, Tokyo) and Yoshioka, T. (Kansai Electric Power Co., Ltd., Osaka).
A/CONF.15/P/1339. 72p.

In the natural-uranium graphite-moderated power reactor, fuel shuffling and continuous charge-discharge on load are necessary in order to obtain a higher burnup factor. It is desirable to predict the change in characteristics of the reactor during operation and the effect of fuel shuffling and continuous charge-discharge. The diffusion equation in which the coefficients vary with position was solved for cylindrical geometry by using the Relay-type Automatic Digital Computer in the Electrotechnical Laboratory. (W.D.M.)

- 14 - 3 CALCULATION OF EFFECT OF FUEL BURNUP ON FUEL AND POISON DISTRIBUTIONS AND ON FLUX DISTRIBUTION IN THE MARINE REACTOR
Hinman, G.
(AECU-3596) General Atomic Div., General Dynamics Corp., San Diego, Calif.
January 29, 1957. 25p. Project 16, Contract AT(04-3)-118. (GAMD-57).

The changes in fuel and poison distributions and the changes in flux shape which accompany the consumption of fuel are studied. The technique employed is a perturbation calculation based on a one-velocity group treatment of the neutrons. The geometry is a spherical core surrounded by an infinite reflector.

The programming forms for the IBM-650 which performs the computations using the Wolontis interpretative system are included. Two sample calculations were worked out using the code described and the results are plotted. (M.H.R.)

14 - 4

DESIGN BASIS FOR THE PEGCPR FUEL ASSEMBLY

Miller, R. S., Burwell, S. B., Davidson, J. K. and O'Leary, W. J. (Nuclear Products-Erco Div.), ACF Industries, Inc., Washington, D. C. In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 54-62

The development of ceramic uranium dioxide fuel is discussed. The design limitations and parameters are given. The requirement for containment of fission products within the fuel elements is explained and the design of the container for the fuel element is reviewed. The helium cooled partially enriched uranium dioxide fuel elements and fuel assemblies are described, and the method of estimating their lifetime is presented. (auth)

14 - 5

ECONOMICS OF ENRICHMENT AND OF THE USE OF PLUTONIUM AND U-233

Franklin, N. L., Hill, J. M. and Stewart, J. C. C. (United Kingdom Atomic Energy Authority, Risley, Lancs. Eng.), and Rennie, C. A. (Atomic Energy Research Establishment, Harwell, Berks, Eng.) A/CONF.15/P/54. 18p.

A standard for comparison, the gas-cooled reactor using natural uranium metal fuel is adopted. A brief survey of the cost structure of power produced from such a reactor is given. The effect of the fuel cycle and the reactor size upon the expected reactivity performance using natural uranium fuel is discussed. The cost structure in the production of enriched uranium is reviewed and a typical cost enrichment curve is deduced for the range 0.5 to 2.0 Co. Typical fuel element costs, inventory costs, and repurchase values are presented. The general principles of the use of plutonium in place of uranium 235 are considered for conditions of total plutonium recycle and for plutonium-natural uranium fuels. The major differences between such systems and systems using enriched uranium are discussed under the headings of: Fuel element fabrication and processing; The choice of spikes or uniform enrichment; The nuclear performance of plutonium as a substitute fuel; and The effect upon reactor operation. These considerations are combined to provide estimates of the relative economic worth of plutonium and uranium 235 enrichment. The general principles of the use of uranium 233 as a fuel are outlined. (auth)

14 - 6

PACKED BED FUEL ASSEMBLY FOR A GAS COOLED HETEROGENEOUS REACTOR
Ristic, M. and Zaric, Z. (Inst. of Nuclear Sciences "Boris Kidric,"
Belgrade).
A/CONF.15/P/492. 6p.

In a gas-cooled heterogeneous reactor the film heat transfer coefficient is low, due to a reasonable pumping power, thus leading to large temperature gradients and thermal stresses in the fuel. From the point of view of a maximum permissible temperature in fuel, there is an upper limit for the exit gas temperature. With the conception of a packed-bed fuel element, which has a better surface-to-volume ratio resulting in a smaller temperature gradient in the fuel, there exists the possibility of improving this situation. A new conception of the fuel element as being a packed bed in the center of a cylindrical channel through and around which cooling gas is flowing axially is discussed. This conception is compared with another one with several fuel rods of small diameter, placed axially in the channel. The comparison shows that the packed-bed fuel element has certain advantages. Thus, for the same pumping power, the same maximum temperature in fuel, and the same heat generation per unit volume, with a packed bed fuel element one obtains a higher temperature gradient along the cooling channel. (auth)

14 - 7

PLUTONIUM RECYCLE IN THE CALDER HALL TYPE REACTOR
Barbieri, L. J., Webster, J. W. and Chow, K. T.
(ASAE-S-8) American-Standard. Atomic Energy Div., Mountain View,
Calif.
January 1, 1958. 116p. Project SIZE-UP. Contract AT(04-3)-109.

The economics and physics of four schemes of plutonium recycle in the Calder Hall type reactor are considered. The four possible schemes are: (1) to blend the Pu produced in a run with fresh natural U for a subsequent run; (2) to alloy the Pu with some diluent metal and fabricate the alloy into high heat-transfer elements more like MTR or PWR-seed type elements and "spike" a subsequent load of fresh natural U elements with these Pu elements; (3) to recycle half the spent U as well as the Pu; and (4) to take advantage of the extra reactivity gained from the recycled Pu to decrease the lattice spacing, thereby increasing the conversion ratio with the hope of increasing the attainable exposure. It was concluded that scheme (1) is the most economical. (W.D.M.)

- 14 - 8 A PRELIMINARY STUDY OF THE TURRET EXPERIMENT. AN OPERATING TEST OF UNCLAD FUEL AT HIGH TEMPERATURE
Hammond, R. P. and Cody, J. P., eds.
(LA-2303) Los Alamos Scientific Lab., New Mex.
March 1959. 91p. Contract W-7405-Eng-36.

A preliminary study has been made of an impregnated graphite reactor to produce helium at 2400°F for process heat uses. The proposed facility will determine the characteristics and problems associated with operation of unclad graphite fuel. Volatile fission products will escape from the fuel elements into the helium stream, which will be continuously purified. Laboratory studies have shown that impregnated graphite will perform satisfactorily at the proposed conditions, and a simple fuel cycle has been developed.

- 14 - 9 PROBLEMS OF FUELLING IN INDUSTRIAL CALDER HALL TYPE REACTORS
Gallie, R. R. and Gott, H. H. (Atomic Energy Research Establishment, Harwell, Berks, Eng.) and Colmer, F. C. W. (Central Electricity Generating Board, London).
A/CONF.15/P/37. 33p.

The two main problems treated are the utilization of the initial fuel charge and the prediction of the statistical relation between bursts and burn-up and to draw the correct conclusions from this prediction. It is clear, however, that this prediction is largely a problem of the accumulation of operating experience and initial estimates are not likely to be precise. Attention is drawn to the practical problems, although the treatment is largely mathematical. (W.D.M.)

- 14 - 10 THEORETICAL ASPECTS OF FUEL CYCLING IN INDUSTRIAL CALDER HALL TYPE REACTORS
Lewis, S. E. (Nuclear Power Plant Co., Ltd., Knutsford, Ches, Eng.) and Lowthian, C. S. (AEI-John Thompson Nuclear Energy Co., Ltd., Knutsford, Ches., Eng).
A/CONF.15/P/311. 23p.

A basic theory of once-through fuel cycles is developed, which considers the variation of reactivity with fuel irradiation. It is shown how values of the fuel life can be obtained for various simple cycles and how these lives are a function of the reactivity which is allocated to life rather than to reactor heat output. The effect of the allocation of reactivity on the total generation cost of electricity is discussed. The concept of an equilibrium state is considered, in which the fuel cycle is so arranged that the over-all reactivity of the reactor does not vary with further irradiation. The reactivity at any specific point within the reactor is, however, a function

of the fuel cycle adopted and it is shown how the resultant perturbations of the flux and specific heat production can be determined. In the approach to the equilibrium state for a reactor loaded with new fuel, a modified fuel cycle can be adopted, in which the fuel is charged at different rates. The conditions for the optimum degree of delayed onset of cycling are established and the effects on the reactivity discussed. (auth)

15. GENERAL and DESCRIPTION

15 - 1

ADVANCES IN THE DESIGN OF GAS-COOLED GRAPHITE-MODERATED POWER REACTORS

Moore, R. V., Kronberger, H. and Grainger, L. (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.). Includes Addendum 1. A/CONF.15/P/312. 23p.

The main engineering features and economic performance of reactors of the improved Calder Hall type, fuelled with natural uranium, are considered. The effect of higher fuel rating on design is discussed; this involves the segregation of the fuel into forms with large surface-to-volume ratios, and the use of slight enrichment. Fuel cycles in relation to this requirement are described. Fuel elements of new materials, in which the present temperature limitation is removed, are under development. It is intended that these will be exploited in the design of British nuclear power stations coming into commission around 1965. The improved performances of these power stations is estimated, and it is shown that the much reduced capital cost per kilowatt will lead to reduced unit cost, even at the lower load factors expected as the use of nuclear power becomes more widespread. Fuel cost can be reduced by the development of fuel elements which will sustain longer burn up. Fissile material in a ceramic form appears to have higher irradiation stability than the metals. These developments, together with the metallurgical development of high temperature canning materials, are discussed. Problems related to the stability and life of the moderator and the safe operation of a highly-rated reactor with plutonium-rich fuel are discussed in detail.
(auth)

15 - 2

AIR COOLED NEUTRONIC REACTOR

Fermi, E. and Szilard, L. (to U. S. Atomic Energy Commission). U. S. Patent 2,836,554. May 27, 1958.

A nuclear reactor of the air-cooled, graphite moderated type is described. The active core consists of a cubicle mass of graphite, approximately 25 feet in each dimension, having horizontal channels of square cross section extending between two of the opposite faces, a plurality of cylindrical uranium slugs disposed in end to end abutting relationship within said channels providing a space in the channels through which air may be circulated, and a cadmium control rod extending within a channel provided in the moderator. Suitable shielding is provided around the core, as are also provided a fuel element

loading and discharge means, and a means to circulate air through the coolant channels through the fuel channels to cool the reactor.

15 - 3 ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM. MONTHLY PROGRESS REPORT
MARCH 1959

- - -

(IDO-28538) Aerojet General Nucleonics, San Ramon, Calif.
April 30, 1959. 49p.

Describes monthly progress on 3 gas-cooled reactors: (1) the water moderated heterogeneous reactor for which major components have been shipped to Idaho, (2) the portable gas-cooled reactor for which nuclear instrumentation specifications are completed and (3) the graphite-moderated homogeneous reactor for which a preliminary core design was continued with special attention to various reflector and coolant passage geometries. Tests were performed to determine the extent to which heat transfer may be controlled in a free convection capsule design.

15 - 4 ATOMIC POWER DEVELOPMENTS

Titterton, E. W.

Elec Eng & Merchandiser, Vol. 32, No. 10, 11, 12, January 1956 p.331-3,
February p. 360-3, March p. 390-4.

Jan: United Kingdom power program; 10-yr program to build 12 nuclear power stations; types of reactor likely to be used.
Feb: Calder Hall power station. Mar: British fast reactor project.

15 - 5 AUSTRALIAN A.E.C. RESEARCH PROGRAMME

Watson-Munro, C. N.

Nuclear Eng, Vol. 1, No. 5, p. 183-5, August 1956.

Objective of Australian Atomic Energy Commission is to carry on feasibility studies of gas cooled impregnated moderator fuel element type and liquid metal cooled reactors; research station at Lucas Heights is to have high flux experimental reactor similar to DIDO reactor; note on facilities under construction.

15 - 6 BEPO. THE LARGER EXPERIMENTAL PILE AT HARWELL

- - -

Atomics, 2:176-81, (June 1951).

A graphite and natural U reactor with a central thermal neutron flux of 10^{12} n/cm²/sec, it contains about 40 tons of U metal in 1 ft x 0.9 in. dia. Al canned "slugs", loaded into 900 horizontal channels to form a cylindrical reacting core in a 26 ft graphite cube. There are 20 slugs per channel, the channels running parallel through the graphite with a 7-1/4 in. square lattice pitch. The core forms a right cylinder of 20 ft dia. and length, the remaining graphite of the cube acting as a neutron reflector. The graphite cube (850 tons) is built up of accurately machined blocks 7-1/4 in. square by 29 in. long. The pile is surrounded by a 7 ft 6 in. thick reinforced concrete biological shield with an inner thermal shield of iron 6 in. thick. Cooling is by 180,000 ft³ of air/min. drawn through by up to 4,1400 h.p. exhausters, which keep the maximum U surface temperature down to 200°C at an operating power of 4MW. Use of exhausters instead of compressors prevents the escape of airborne activity, the exhaust being expelled via a 200 ft stack, after filtering. Ingoing air is also filtered. Control is by 4 horizontal B₄C-filled steel tubes 2 in. dia. and 10 similar shut-off rods operating vertically by compressed air. Experimental facilities include 2 thermal columns and a number of experimental holes through the pile.

15 - 7

THE BR-1 PILE. PART I. DESCRIPTION AND USE. PART II. PHYSICAL CHARACTERISTICS

Neve de Mevergnies, M., Fairclough, C., Stiennon, G. and Leonard, F. (Centre d'Etude de l'Energie Nucleaire, Brussels).
A/CONF.15/P/1673. 62p.

The BR-1 reactor which was built at Mol (Belgium) is described in detail. This is a graphite-moderated reactor which uses natural uranium. Cooling is by air, using an open circuit. It operates at a power of 5 Mw. The experimental possibilities and program for use of the reactor are reviewed, along with the special research equipment. An experiment to measure the spectral distribution of slow neutrons in the uranium-graphite lattice is described. The transfer function of the reactor was measured in order to determine the mean lifetime of the neutrons in the pile. The diffusion length of thermal neutrons in the core was measured. (W.D.M.)

15 - 8

BERKELEY NUCLEAR POWER STATION

Nuclear Eng, Vol. 2, No. 12, p. 96-102, March 1957.

Details of A.E.I.—John Thompson design; general layout; pressure vessel; shielding; gas cooling system; heat exchangers; fuel element changing; control; burst slug detection; construction of reactors; erection of heat exchangers and associated works.

15 - 9

THE BERKELEY POWER STATION

Ghalib, S. A. and Southwood, J. R. M. (John Thompson Nuclear Energy Co. Ltd., Knutsford, Ches., Eng.).
A/CONF.15/P/264. 43p.

The major design features of the Berkeley Power Station are described, and the considerations leading to their choice are discussed. Detailed descriptions are given for the various reactor components, and the operational characteristics are discussed. (W.D.M.)

15 - 10

BRADWELL NUCLEAR POWER STATION

Vaughan, R. D. and Anderson, E. (Nuclear Power Plant Co., Ltd., Knutsford, Eng.)
A/CONF.15/P/263. 34p.

Descriptions of the site, reactor, and auxiliaries of the Bradwell nuclear power station are presented. Theoretical and physical features of the core are explained with particular reference to the advanced type of fuel element developed especially for this design. The boilers, gas circuits, and reactor vessel are described along with the control system and turbo-generator section of the plant. (W.D.M.)

15 - 11

BRADWELL NUCLEAR POWER STATION

- - -
Nuclear Power 2, 410-13 (1957) October.

A detailed description of the Bradwell Nuclear Power Station is given. The principal nuclear and engineering data, the charging system, the safety features, shielding, power generation system, and control system details are summarized. A 2 by 3 ft colored cutaway drawing of the installation is included. (M.H.R.)

15 - 12

THE BROOKHAVEN REACTOR

Fox, M.

In International Conference On The Peaceful Uses Of Atomic Energy, Geneva, 1955, Proceedings. Vol. 2, Physics: Research Reactors. New York, United Nations, 1956. p. 353-371.

The discussion includes design concepts, design problems and their treatment, control system, fuel rod leak detection methods, the reactor shield, experimental facilities, the operating schedule and philosophy, the operations staff, startup and shutdown procedures and their effect on fuel element life, measured flux pattern and spectra, list of experimental

equipment, some effects produced in graphite by neutron irradiation in the reactor, an experiment to determine behavior of the reactor upon loss of coolant, some aspects of the history of operations, construction costs, and plans for refueling the reactor.

15 - 13 CALDER HALL

- - -

Nuclear Eng. 1, 263-308 (1956) October.

The world's first large scale Nuclear Power Station at Calder Hall is discussed. The design and construction of the reactor, pressure vessel, fuel charge and discharging gear, control rods, burst slug detention gear, heat exchangers, coolant gas systems, graphite placing and civil engineering problems are included. (F.S.)

15 - 14 CALDER HALL CONFERENCE

- - -

Nucleonics 14, No. 12, S10-S32 (1956) December.

A detailed cutaway of the Calder Hall Reactor is presented. The initial operation, reactor physics, metallurgical development, heat transfer, blower seals, burst-slug detection, fuel handling, and control rod data are given. (W.L.H.)

15 - 15 CALDER HALL

- - -

Nuclear Power, Vol. 1, No. 8, p. 316-38, December 1956. See also Brit Nuclear Energy Conference--J Vol. 2, No. 2, p. 41-228, 2 April 1957. 2 supp plates, (discussion) No. 3, p. 233-91, July.

Future Developments, R. V. MOORE: Basic Design, G. PACKMAN, B. CUTTS: Design Study, R. V. MOORE, B. L. GOODLET: Plant, R. V. MOORE: Fuel Element Heat Transfer, P. FORTESCUE, W. B. HALL: Experimental Physics, P. W. MUMMERY: Shield Design, C. C. HORTON, W. BONSALL: Early Metallurgical Problems, R. A. V. HUDDLE, L. M. WYATT: Metallurgical Developments, L. GRAINGER, A. B. McINTOSH: Reactor Vessel, G. BROWN, M. J. NOONE, R. F. BISHOP: Fuel Handling, K. H. DENT, G. W. GROSSMITH: Important Plant Items, A. T. BOWDEN, G. H. MARTIN: Steam Cycle Analysis, W. R. WOOTTON: Heat Exchangers, H. MORRIS, W. R. WOOTTON: Control Equipment, S. A. GHALIB, J. H. BOWEN: Detection of Faulty Fuel Elements, E. LONG, J. M. LAITHWAITE, K. W. CUNNINGHAM: Control and Instrumentation, R. J. COX, K. R. SANDIFORD: System Control and Protection, E. ANDERSON, J. H. BOWEN: Calder Hall Type in Power Production, C. HINTON.

15 - 16 CALDER HALL NUCLEAR POWER STATION

Frost, K. J.

Soc Engrs--J & Trans, Vol. 47, No. 4, p. 89-110, October-December 1956.

Initial problems encountered in Calder Hall project and way in which these were overcome; general description of reactor; shutdown system is incorporated in control rod system; by using magnetic clutch arrangement in support for control rods they can be dropped into core by demagnetizing clutch; special precautions taken to ensure that fission product contamination does not occur throughout coolant system.

15 - 17 CALDER HALL POWER STATION

Engineer, Vol. 200, No. 5209, p. 766-8, November 25, 1955. see also Indus Chemist, Vol. 31, No. 371, p. 598-601, December 1955. Eag & Boiler House, Rev. Vol. 71, No. 1, p. 4-10, January 1956. Steam Engr., Vol. 25, No. 293, p. 162-5, February 1956.

Site adjoins Windscale plutonium factory; two stations. "A" and "B", of similar output are being built "in tandem" on same longitudinal axis; each reactor is graphite moderated, pressurized pile, cooled by carbon dioxide in closed cycle; four Parsons turboalternators each having rating of 23 Mw at 11 kv and 3000 rpm; turbines are 2-cyl reaction machines.

15 - 18 CHAIN REACTING SYSTEM

Fermi, E. and Leverett, M. C. (to U. S. Atomic Energy Commission). U. S. Patent 2,837,477. June 3, 1958.

A nuclear reactor of the gas-cooled, graphite-moderated type is described. In this design, graphite blocks are arranged in a substantially cylindrical lattice having vertically oriented coolant channels in which uranium fuel elements having through passages are disposed. The active lattice is contained within a hollow body, such as a steel shell, which, in turn, is surrounded by water and concrete shields. Helium is used as the primary coolant and is circulated under pressure through the coolant channels and fuel elements. The helium is then conveyed to heat exchangers, where its heat is used to produce steam for driving a prime mover, thence to filtering means where radioactive impurities are removed. From the filtering means the helium passes to a compressor and an after cooler and is ultimately returned to the reactor for recirculation. Control and safety rods are provided to stabilize or stop the reaction. A space is provided between the graphite lattice and the internal walls of the shell to

allow for thermal expansion of the lattice during operation. This space is filled with a resilient packing, such as asbestos, to prevent the passage of helium.

- 15 - 19 CHAPELCROSS POWER STATION
Knowles, P.
Nuclear Power 3, 304-12 (1958) July.

The Chapelcross nuclear power station will contain four 180 Mw gas-cooled, graphite-moderated reactors of the Calder Hall design and eight 23 Mw turboalternators. The primary purpose of the works is the production of plutonium but it will have a net electrical output of about 140 Mw. The differences in design between Chapelcross and Calder Hall are emphasized. A description of the site and the construction progress is given. (W.D.M.)

- 15 - 20 THE CHINON NUCLEAR POWER PLANT DIVISIONS EDF 1 AND EDF 2
Roux, M. and Bienvenu, M. (Electricite' de France, Paris).
A/CONF.15/P/1135. 100p.

The characteristics and the over-all plan of the EDF-1 and -2 plants are described. The construction and work programs are reviewed. Both plants are natural uranium graphite reactors cooled by CO₂ under 355 lb/in.² pressure. The EDF-1 is a 82 Mwe plant using 140 tons of U with the steam pressure lower than the CO₂ pressure. The EDF-2 is a 200 Mwe plant using 250 tons of natural U with the steam pressure everywhere greater than the CO₂ pressure. (W.D.M.)

- 15 - 21 THE CHINON NUCLEAR POWER PLANT, DIVISIONS EDF 1 AND EDF 2
Roux, M. and Bienvenu, M. (Electricite' de France, Paris).
A/CONF.15/P/1135. (Add. 1), 29p.

Illustrations and figures are presented that were omitted from the English text of the report. (W.D.M.)

- 15 - 22 CURRENT RE-DESIGNS OF CALDER HALL
Cunningham, J. B. W. Includes Appendix: LEAKAGE OF CO₂ FROM THE PRESSURE CIRCUIT OF THE CALDER HALL REACTORS. Disney, H. V. (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.).
A/CONF.15/P/73. 19p.

The first re-design of the Calder Hall reactors was carried out by industrial firms in Britain for U.K. electricity

authorities. The U.K. Atomic Energy Authority collaborated with the electricity authorities in writing the specification for these stations and in assessing the designs. The method adopted for carrying out the assessment is described. It involved a detailed consideration of each design from a number of different aspects, namely: performance, safety, fuel elements, pressure circuit, control and instrumentation, graphite structure, burst cartridge detection, and civil engineering. This re-design, and later ones now current, led to a very great advance in performance and economy over Calder Hall. The various factors responsible for the improvement are discussed, including increases in the diameter and working pressure, and improved internal geometry of the main pressure vessel; increased rating and temperature of the fuel and new heat transfer surfaces for the fuel elements. Modifications to the design which have arisen from consideration of postulated fault conditions, and the development of major components such as main gas valves and bellows are considered. Work is reported on the leak tightness of the Calder Hall pressure circuit, and on the principles of design to be applied to similar circuits. (auth)

15 - 23 DESCRIPTION OF REACTORS G2-G3
 - - - (France-Atome.)
 A/CONF.15/P/1134. 50p.

The reactor core includes most of G-1 reactor parts including horizontal channels, graphite parallelepipedic bricks stacking, and steel thermal shield. The cooling is provided with CO₂ under a 15 atmospheres pressure. This pressure is kept steady in a press-stressed concrete packing-case in the form of a horizontally shaped cylinder. Packing-case tightness is provided by a 30 mm iron plate connected with the inner wall of concrete. One of the reactor's special characteristics is the possibility of loading and unloading while operating. On the loading side, barrel-locks, each weighing 50 tons, allow new cans, at a pressure of 15 atmospheres, to pass. The cans process almost in a steady way through the channel, and finally drop down through bent spouts, then through spiral toboggans into a new lock. The cooling CO₂ flow is provided with 3 turbo-bellows; these are actuated by average pressure steam, obtained from exchangers. Every reactor supplies 4 exchangers. The secondary cycle is standard and contains 3 stages. Steam can be condensed in the event of a group turbo-generator stopping, with no modification for the normal operating conditions of the reactor. Auxiliary circuits have to assure the continuous purifying of cooling CO₂, its storage, and drain. Forty-nine boron carbide rods are used to control the operating power levels of reactor. The regulating system has brought about difficult problems; however, experimental examination, while operating, will solve them. (auth)

- 15 - 24 DEVELOPMENT OF ATOMIC POWER PLANTS IN GREAT BRITAIN
Walter, L.
S African Min & Eng J, Vol. 65, pt 2, No. 3217, p. 189, 191, 193,
October 9, 1954.

Problem of finding substitute for solid, liquid or gaseous fuels; shortage of fuels in Great Britain will be met by atomic power stations; construction of experimental atomic power station at Calder Hall adjacent to existent Windscale plutonium producing factory at Sellafield, Cumberland; arrangement of reactor under construction.

- 15 - 25 DEVELOPMENT OF NUCLEAR POWER
Titterton, E. W.
Instn Engrs, Australia--J, Vol. 28, No. 9, p. 243-7, September 1956.

Description of Calder Hall station and Dounreay fast reactor station, Great Britain; development of atomic power in Australia.

- 15 - 26 DEVELOPMENT OF SECOND FRENCH REACTOR
Kowarski, L.
Nucleonics, Vol. 12, No. 8, p. 8-11, August 1954.

Objectives underlying construction of Saclay reactor as step toward solving problem of nuclear power on industrial scale; factors considered in choice of reactor design, cooling, etc; various piping layouts required; construction, operation and testing of reactor; results obtained to date and future prospects.

- 15 - 27 EXPERIMENTAL REACTOR SYSTEMS
Jackson, R. F.
Nuclear Eng, Vol. 1, No. 1, 2, April 1956, p. 6-10, May, p. 72-7.

Use of exponential piles, critical assemblies, and zero energy reactors as logical steps in development of high powered reactors, particular reference being made to Britain's Harwell reactors. Apr: Details and cutaway diagrams of DIMPLE and ZEPHYR zero energy reactors. May: Details and cutaway diagrams of ZEUS reactor; some features of GLEEP, BEPO and DIDO reactors.

- 15 - 28 GLEEP: DESIGN, CONSTRUCTION AND USE
Colmer, F. C. W. and Littler, D. J.
Nucleonics, 8:3-12 (January 1951).

GLEEP is a graphite-moderated natural U reactor having the form of a right octagonal prism lying on one side, with a cylindrical core 5.24m in length and 2.86m radius. Twelve tons of U metal in the form of Al-sprayed rods 12 in. long by 0.9 in. dia. are loaded in the control region of the core and 21 tons of dioxide, pressed into pellets 1.6 in. dia. and canned in Al are loaded in the outer channels. There is a 5 ft square 7 ft long graphite thermal column, extending through the concrete shield on one side. Limited air cooling is provided and allows a maximum power of 100 kw. Five Cd control rods and 6 Cd shutoff rods are provided, the reproduction constant of the pile with these fully removed is 1.002. Resistance thermometers measure graphite and U rod temperatures, and a shut-down occurs if the maximum recorded U surface temperature exceeds 60°C. Seven experimental channels through the reactor are provided and a maximum flux of 3.0×10^{10} n/cm²/sec is available at 100 kW. A mechanical oscillator is installed for swinging samples in the pile in order to determine thermal neutron cross-sections (by observing the magnitude of the modulation of pile power). The use of a temperature-controlled wire-wound resistance of 10^8 ohms in series with the BF₃ chambers has made it possible to hold the neutron density at very accurately reproducible levels and hence to provide standard fluxes in the range 10^6 - 10^8 n/cm²/sec.

- 15 - 29 GAS COOLED NUCLEAR REACTOR IN WHICH THE MODERATING SYSTEM IS SEPARATED FROM THE COOLANT SYSTEM BY A BARRIER LAYER
- - - (To Siemens-Schuckertwerke AG., Berlin u. Erlangen, Ger.)
German Patent DAS 1031 901. Atompraxis 5, 42(1959). (In German)

A liquid barrier would be provided between moderator and coolant in gas-cooled, liquid-moderated reactors in which the moderating system and coolant system have pressure equalization. As barrier substance, a deuterated organic liquid with moderating or reflector properties, which floats on the moderator liquid, would be used. In this manner, fission products from defective fuel elements would not pass from the coolant system to the moderator.

- 15 - 30 GAS COOLED NUCLEAR REACTORS
Long, E. and Rodwell, W. (to U. S. Atomic Energy Commission).
U. S. Patent 2,838,451. June 10, 1958.

A gas-cooled nuclear reactor consisting of a graphite reacting core and reflector structure supported in a containing vessel is described. A gas sealing means is included for sealing

between the walls of the graphite structure and containing vessel to prevent the gas coolant by-passing the reacting core. The reacting core is a multi-sided right prismatic structure having a pair of parallel slots around its periphery. The containing vessel is cylindrical and has a rib on its internal surface which supports two continuous ring shaped flexible web members with their radially innermost ends in sealing engagement within the radially outermost portion of the slots. The core structure is supported on ball bearings. This design permits thermal expansion of the core structure and vessel while maintaining a peripheral seal between the two elements.

15 - 31 GAS-COOLED PROJECTS

- - -

Nucleonics 17, No. 2, 52-3 (1959) February.

Information is presented on the design and operating characteristics of facilities for testing experimental reactor cores and gas-turbines-compressor sets. These facilities include the Gas Cooled Reactor Experiment (GCRE) being built at the National Reactor Testing Station and the closed-cycle Gas-Turbine Testing Facility (GTTF) that is under construction at the Army Engineer Research and Development Laboratories, Fort Belvoir. (J.H.M.)

15 - 32 GAS-COOLED REACTORS

In Dietrich, J. R. and Zinn, W. H. eds.,
Solid Fuel Reactors. Reading, Mass., Addison-Wesley, 1958.
Chapter 6, p. 475-666.

This chapter attempts to consolidate the information that has been developed for gas-cooled reactors, which have received serious attention in the United States but which are not yet represented by operating prototype plants. Heat removal from single and multiple channels is discussed. The discussion also includes the relative performance of various gases and the choice of a coolant. Fuels and fuel-element materials are covered, including uranium metal, uranium metal jacket materials, and uranium dioxide. Two types of reactors, by moderator, are discussed -- the graphite-moderated reactor and the D₂O-moderated reactor. Included are the general characteristics, neutron-physics characteristics, natural uranium design studies, and enriched uranium design studies of graphite-moderated reactors. Under D₂O-moderated reactors topics discussed are fundamental characteristic of the concept, design concepts, performance of the prototype, and performance of a large reactor. (W.H.R.)

- 15 - 33 GENERAL SPECIFICATIONS AND ORIGINAL ASPECTS OF REACTORS G2 AND G3
Pascal, (J.) Horowitz, (J.) Bussac, Joatton, de Lagge de Meux, and
Martin (Commissariat à l'Energie Atomique, Paris).
A/CONF.15/P/1133. 15p.

The objectives of G2 and G3 were the production of plutonium along with a significant amount of electricity. The initial specifications were chosen as a direct result of experience gained from G1 and EL2. The essential requirements were to maintain the pressure of the cooling gas slightly above that of the steam, the loading and unloading of fuel under pressure and operation, horizontal arrangement for the fuel elements, and prestressed concrete for the pressure vessel. The prestressed concrete allowed for a high operating pressure and improved performance. (W.D.M.)

- 15 - 34 GIORNATE DELL'ENERGIA NUCLEARE

Energia Nucleare, Vol. 5, No. 12, (December 1958). Conference on
Nuclear energy held in Milan, Italy, December 1958.

EDF-1 and EDF-2 nuclear power plants of Electricité de France,
J. P. Roux, p. 5-14. (In French). Latina nuclear power station
and nuclear energy development in Italy, G. Martinoli, p. 15-24.
Some considerations on gas cooled reactors, V. Benzi, A. Calori,
p. 142-51.

- 15 - 35 THE GRAPHITE-MODERATED, GAS-COOLED PILE AND ITS PLACE IN POWER
PRODUCTION
Hinton, C.
In International Conference on the Peaceful Uses of Atomic Energy,
Geneva, 1955, Proceedings. Vol. 3, Power Reactors. New York,
United Nations, 1956. p. 322-329.

Although graphite-moderated water-cooled piles which are not inherently stable were built at Hanford for plutonium production, we in Great Britain did not follow suit because of the large safety distances involved. Instead we constructed two graphite-moderated air-cooled piles which are inherently stable to produce fissile material, having first decided that reasonably economic operation could be attained by flattening of the neutron flux across the core.

In the two piles at Windscale which have been described elsewhere all the heat is wasted, but in the two graphite-moderated gas-cooled piles, at nearby Calder Hall, the heat of fission is utilized to raise steam which drives turbo-alternators for the production of electricity. The graphite moderator is contained in a cylindrical steel pressure vessel which is surrounded by a steel thermal shield and by a concrete biological shield.

The fuel elements are accommodated in vertical channels in the graphite while the charge and discharge arrangements along with the control and shut-off gear are situated on the pile vault roof. The hot CO₂ coolant from the pressure vessel is passed thru four heat exchangers before being boosted by the blowers back to the pressure vessel. Steam is raised at two pressures in the heat exchangers which contain economiser, evaporator and superheater sections, and then it is led away to the generator house to drive the turbo-alternator sets. These sets have a full load capacity of 23 megawatts each at 11 KV from which power is exported to the Central Electricity authority's grid distribution lines. A closed circuit water cooling system incorporating twin cooling towers is employed for the steam condensers.

The recent White Paper published by the British Government shows that when a reasonable allowance is made for the value of the plutonium by-product from the graphite-moderated gas-cooled piles, electrical power can be generated for 0.6-pence per unit. It should be possible to reduce this cost in future thermal reactor designs by progressing with larger and thicker pressure vessels or by finding more suitable coolants. The use of a slightly enriched uranium charge may also lead to economies while there is also scope for improved and cheaper designs in the pure engineering field.

The construction of the graphite-moderated gas-cooled reactor is the basis of the British program for industrial power which in the decade until 1965 should realize a total nuclear power generating capacity of 1400-1800 megawatts.

It is useful to think of nuclear reactors in an ascending scale of the specific heat rating of their fuel elements and we in Britain are attacking the two ends of this scale. At one end lies the Calder Hall thermal reactor, the slow speed reciprocating engine of the reactor world, while at the other end lies the Dounreay fast reactor, the high speed gas turbine of this world, but it would be a mistake to consider that the former has only a limited future.

- 15 - 36 THE HIGH-TEMPERATURE GAS-COOLED REACTOR
 - - -
Atomic Energy Rev. 1, 115(1958) September.

Not looked at.

- 15 - 37 THE HIGH TEMPERATURE ZERO ENERGY REACTOR "ZENITH."
 Mitchell, K. J. and Geary, R. A. (General Electric Co., Ltd., Erith, Kent, Eng.).
 A/CONF.15/P/1463. 12p.

"Zenith" is a zero energy reactor in which various types of experiments at high temperatures may be carried out in order to

obtain data for high temperature gas cooled reactors. In order to produce the expected mean temperatures of core and reflector of larger reactors the circulating coolant gas is electrically heated prior to entering the core and the waste heat is rejected from the system in a gas to water heat exchanger. It is expected to go critical in early 1959. Components described include core and reflector, pressure vessel, gas circuit and blower, control and instrumentation, shielding and containment, charge-discharge, and ventilation plant. (W.D.M.)

- 15 - 38 HINKLEY POINT. THE WORLD'S REACTORS. NO. 19. HINKLEY POINT POWER STATION

- - -

Nuclear Eng. 3, 286-6a(1958) July.

Details for the gas-cooled two-reactor Hinkley Point Power Station are given. A large schematic drawing of one of the reactor units is presented. (W.D.M.)

- 15 - 39 HUNTERSTON GENERATING STATION--DESCRIPTION OF NUCLEAR AND MECHANICAL PLANT

Oakes, J. S.

Instn Mun Engrs--J, Vol. 85, no. 11, p. 391-5, 400. November 1958.

Station is comprised of two reactors each associated with 8 steam-raising units, 3 alternators, ancillaries, and steam-dumping equipment; coolant CO₂ gas is circulated through reactor, and heat from gas is extracted in steam-raising units to form steam for driving turbogenerators; 70-ft diam pressure vessel is contained within thick biological wall of concrete; fuel elements, charge machine, control rods, and turbines described; 360 Mw of electricity will be produced.

- 15 - 40 IMPROVEMENTS IN OR RELATING TO NUCLEAR REACTORS

Fenning, F. W. and Jackson, R. F. (to United Kingdom Atomic Energy Authority).

British Patent 779,134. July 17, 1957. Nuclear Power 2, 394(1957) September.

Very long and detailed, this is the full specification for Harwell's famous air-cooled reactor BEPO which first went critical in 1948. Since the date of application is nearly four years after the pile first went into operation the specification represents it in actuality, these being only one claim. (auth)

15 - 41

IMPROVEMENTS IN OR RELATING TO NUCLEAR REACTORS

Moore, R. V., Dent, K. H. and Bowen, J. H. (to United Kingdom Atomic Energy Authority).

British Patent 781,648. August 21, 1957. Nuclear Power 2, 486(1957) November

This is the basic patent for the PIPPA design study for a natural-uranium, gas-cooled graphite-moderated power reactor that later became Calder Hall. It apparently covers a core with vertical fuel element channels arranged on a regular lattice in groups with a control rod in the center of each group. The whole core is contained in a pressure vessel and access to the fuel and control rods is from above through standpipes. Heat exchanger and gas circulation systems are described in the specification, as is the burst slug detection system. An interesting point covered is the design of graphite blocks to permit Wigner growth at the same time maintaining the lattice geometry. As is well known, the final Calder Hall design was rather different in some respects from PIPPA. In particular the patent specification covers ledges in the graphite fuel channels for separate support of the fuel elements: at Calder this was abandoned in place of straightforward stacking. (auth)

15 - 42

INFORMATION MEETING ON GAS-COOLED POWER REACTORS, OAK RIDGE NATIONAL LABORATORY, OCTOBER 21-22, 1958

- - -

(TID-7564) Oak Ridge National Lab., Tenn. 376p.

This meeting is one of a series of Civilian Power Reactor Conferences and was held coincident with an AEC invitation to industry to bid on the construction of a gas-cooled facility. Papers are presented on design studies, hazards, components, costs, materials, and design concepts for specific reactors. (W.D.M.)

15 - 43

LES INSTALLATIONS DE PRODUCTION D'ÉNERGIE DE MARCOULE ET LA CENTRALE NUCLEAIRE DE CHINON

Teste, Y.

Société des Ingénieurs Civils de France--Memoires, Vol. 110, No. 2, p. 61-75, March-April 1957.

Installations at Marcoule power plant and nuclear power plant at Chinon, France; description of design and functioning of nuclear reactor, which started operations in September 1956; conversion of surplus heat into electric power; French atomic planning until 1975 is outlined.

- 15 - 44 JAMES FORREST LECTURE, 1958—FURTHER DEVELOPMENT OF UNITED KINGDOM
NUCLEAR POWER PROGRAMME
Cockcroft, J.
Instn Civ Engrs—Proc, Vol. 10, p. 437-52, August 1958.

Sizes of reactors have been increased in last 3 yr; experiences at Calder Hall reactor; radiation-induced creep by passage of fission fragments through uranium; experiments on carbon monoxide level; release of stored energy on Harwell's BEPU reactor; sodium graphite, gas cooled graphite moderated, and heavy water moderators; use of sintered uranium oxide; achievements in fusion field; Zeta reactor.

- 15 - 45 DE KERNRECKTOREN VOOR DE VOORTBRENGING VAN ENERGIE
Vervoort, L.
Technisch Wetenschappelijk Tijdschrift, Vol. 27, No. 2, p. 25-38,
February 1958.

Elements of nuclear physics and fundamental principles of nuclear core reactors; high temperature and structural materials for experimental and energy producing reactors using heavy water, graphite and other moderators; description of reactors at Calder Hall, England; Shippingport, Pa, and in USSR; it is shown that present research is directed toward fast reactor systems using uranium, and thermal systems using thorium.

- 15 - 46 LAUNCHING A PROGRAMME OF PRESSURIZED GAS-COOLED REACTORS
Horowitz, J. and Yvon, J. (Commissariat à l'Energie Atomique, Paris).
A/CONF.15/P/343, 8p.

The natural uranium and plutonium system of power generation development is reviewed along with the use of compressed gas as a primary fluid, The EL-2, a natural uranium heavy water and compressed gas pile, and the G-1, a natural uranium graphite and atmospheric air pile, are examined in relation to their contribution to development. A number of factors governing investment costs and an economically acceptable utilization of fuels are reviewed in relation to gas cooled, graphite-moderated, and heavy water reactors. (J.R.D.)

- 15 - 47 LECTURES ON NUCLEAR SCIENCE AND ENGINEERING, NOVEMBER 16, 1948 to
FEBRUARY 1, 1949

(BNL-24) (Del.) Brookhaven National Lab., Upton, N. Y. 153p.

Lectures prepared for a survey course on reactor theory, design,

operation, and experimentation are presented. General reviews of neutron and fission physics are made and the Brookhaven Reactor design, engineering, and operation are discussed. (D.E.B.)

15 - 48 NUCLEAR POWER

Rotblat, J.

Inst Fuel—J, Vol. 26, No. 151, p. 109-14, August 1953.

Types of reactors; feature of British Experimental Pile (BEPO); water boiler "HYPO"; reactors for power production; breeder reactors; cost of nuclear power.

15 - 49 NUCLEAR REACTOR

Moore, R. V., Bowen, J. H. and Dent, K. H. (to U. S. Atomic Energy Commission).

U. S. Patent 2,863,815. December 9, 1958.

A heterogeneous, natural uranium fueled, solid moderated, gas cooled reactor is described, in which the fuel elements are in the form of elongated rods and are disposed within vertical coolant channels in the moderator symmetrically arranged as a regular lattice in groups. This reactor employs control rods which operate in vertical channels in the moderator so that each control rod is centered in one of the fuel element groups. The reactor is enclosed in a pressure vessel which is provided with access holes at the top to facilitate loading and unloading of the fuel elements, control rods and control rod driving devices.

15 - 50 NUCLEAR REACTOR TECHNOLOGY

de Bruyn, H., Bustraan, M., et al.

Gravenhage, Netherlands, Reactor Centrum Nederland, 1958. 355p. (In English and Dutch)

A series of articles on nuclear energy are presented. A survey of reactor theory is given, and three reasons for a disturbance of the critical state of a reactor are discussed. The physical processes which form the principles for the calculations on reactor shielding are surveyed. The reactors of the U.K. power program are described. An outline of the implications of physics on reactor fueling is given. The nuclear reactor in chemical technology is discussed. The influence of the coolant pressure and the pressure drop over the steam generators with respect to the heat exchange surface needed to transfer a certain amount of heat is discussed. Reactor vessels for gas-cooled reactors are described. The operation of gas-cooled,

graphite-moderated reactors is surveyed. The results of six-years' operation of JEEP are presented. The Dutch Materials Testing and Research Reactor is described in detail. A brief description is given of the swimming pool reactor to be built at the University of Delft. A brief historical review is given of the operation of graphite research reactors, and the Belgian Reactor BR-1 is described in detail. The application of research reactors to chemistry is discussed. (J.S.R.)

15 - 51

NUCLEAR REACTORS

Long, E. and Ashley, J. W. (to U. S. Atomic Energy Commission).
U. S. Patent 2,864,759. December 16, 1958.

A graphite moderator structure is described for a gas-cooled nuclear reactor having a vertical orientation wherein the structure is physically stable with regard to dimensional changes due to Wigner growth properties of the graphite, and leakage of coolant gas along spaces in the structure is reduced. The structure is comprised of stacks of uniform right prismatic graphite blocks positioned in layers extending in the direction of the lengths of the blocks, the adjacent end faces of the blocks being separated by pairs of tiles. The blocks and tiles have central bores which are in alignment when assembled and are provided with cooperating keys and keyways for physical stability.

15 - 52

NUCLEAR REACTORS AND POWER PRODUCTION

Hinton, C.

Chartered Mech Engr, Vol. 1, No. 5, p 235-51, May 1954, see also Engineer, Vol. 197, No. 5119, 5120, p. 357-60, March 5, 1954. p. 374-7, March 12. Engineering, Vol. 177, No. 4597, p. 313-5, March 5, 1954. Indus Chemist, Vol. 30, No. 351, p. 179-80, April 1954.

Design and construction, with particular reference to British experimental pile (BEPO) built at Harwell in 1948; features of Windscale pile for plutonium production; limitations of temperatures and safety; alternative types of reactor for power generation; prospects for large scale power production for civil and industrial use; underlying theory of atomic energy simply explained. James Clayton Lecture before Instn Mech Engrs.

15 - 53

NUCLEAR REACTORS FOR GENERATION OF POWER

Iliffe, C. E.

North East Coast Instn Engrs & Shipbldrs—Trans, Vol. 72, pt 4, p. 183-204, February 1956. (discussion) pt 7 May-June p. D81-8.

Combinations of moderator and coolant for heterogeneous, thermal neutron type; examples and advantages of heavy water and light water reactors of boiling or pressurized type, water and gas cooled graphite reactors, and sodium cooled graphite reactor; factors governing upper temperature at which heat is transferred; cost comparisons for selected reactors and for power produced from hydrocarbon fuel.

- 15 - 54 PHYSICAL, ENGINEERING AND CHEMICAL PROBLEMS OF NUCLEAR ENERGY
Naude, S. M.
S African Instn Mech Engrs—J, Vol. 4, No. 2, p. 41-50, September 1954.
(discussion) 50-4.

Principles and problems of nuclear energy development as they bear on power production; features of British reactor BEPO at Harwell, England, and of Daniels atomic power plant; technological problems of power production; need for nuclear power source in relation to world's fuel reserves; implications regarding use of South Africa's uranium supply.

- 15 - 55 PLACE OF CALDER HALL REACTOR IN NUCLEAR POWER GENERATION
Hinton, C.
Engineer, Vol. 202, No. 5261, p. 726-8, November 23, 1956.

Graphite moderated gas cooled reactor placed in its correct perspective in atomic energy field. Lecture at Brit Nuclear Energy Conference.

- 15 - 56 PROGRESS ON MATERIALS—KEY TO GAS-CYCLE REACTOR POTENTIAL
- - -
Nucleonics, Vol. 14, No. 3, 42-4 (March 1956).

A brief review of present ideas on gas-cooled power reactors, including comparison data for four proposed systems with electrical outputs up to 60 MW. These reactors have high efficiency and require relatively little space.

- 15 - 57 PROPOSED BROOKHAVEN MEDICAL RESEARCH REACTOR
Robertson, J. S., Stickley, E., Bond, V. P. and Farr, L. E.
Nucleonics, Vol. 13, No. 12, p. 64, 66-8, December 1955.

Attainment of adequate depth doses of irradiation within desirably short time periods for medical and biological studies requires thermal-neutron beam having flux of 10^{10} to 10^{11} n/cm²/sec; requirements should be met by reactor operating at up to 1 Mw power using "swimming pool" type fuel elements and air cooled graphite reflector moderator.

- 15 - 58 QUELQUES ASPECTS TECHNIQUES DES REACTEURS NUCLEAIRES
Symon, E.
Société Royale Belge des Ingenieurs et Industriels, No. 2 p. 75-84,
February 1955.

Some technical aspects of nuclear reactors; role of reactor; fuels; requirements of moderator; homogeneous and heterogeneous reactors; cooling methods; control and safety; first Belgian reactor described.

- 15 - 59 REACTEUR G-1 ET SA CENTRALE DE RECUPERATION D'ENERGIE
Pascal, M. and Roux, J. P.
Technique Moderne, Vol. 49, No. 4, 6, p.21-6, April 1957, p.249-56,
June.

G 1 reactor and its energy recovery plant. Apr: Marcoules reactor uses natural uranium fuel, has graphite moderator, and is air cooled by open circuit; characteristics of reactor, materials handling devices, and control and checking equipment. June: Reactor liberates 35 MM calories per hr; heat recovery power plant in final stage of circuit produces energy from part of this heat in form of air at 220 C delivered at rate of 566 lb per sec; net power is 5100 kw.

- 15 - 60 REACTORS AND POWER PRODUCTION AT MARCOULE
- - -
Nuclear Eng. Vol. 1, No. 1, p. 22-4, April 1956.

Features of Electricite de France's three reactors at Marcoule now designed to generate electric power as well as produce plutonium; details of G1 and G2 piles with central entry cooling are given with brief description of plutonium production plant; for G1, output will be only 5000 k but with G2 and G3, it is hoped to attain net power output of 20,000 k from 100,000 k of heat generated.

- 15 - 61 THE REACTORS OF THE FRENCH PROGRAM
Horowitz, J. (Commissariat a l'Energie Atomique, Paris).
In 4^a Rassegna Internazionale Elettronica e Nucleare. Atti del Congresso Scientifico. Volume I. Sezione Nucleare, Giugno 1957.
(4th International Electronic and Nuclear Review. Acta of the Scientific Congress. Volume 1, Nuclear Section, June 1957.)
Rome, Segret, Comitato Nazionale delle Ricerche Nucleari, 1958.
p. 127-38. (In French)

The reactors of the French nuclear power program include G-1, G-2, G-3, and EDF-1, and each is discussed. Some prototypes which are being planned are briefly described.

15 - 62

- - -

REACTORS OF THE WORLD. FIRST SERIES

New York, Simmons-Boardman Publishing Corporation, 1958. 29p.

Cutaway drawings and specifications are presented of the NRX, Bradwell-on-Sea, Dounreay, DIDO, Calder Hall, LIDO, ZEUS, DIMPLE, BEPO, The Russian 5 Mw power, the EBWR, and the CP5 reactors. (J.S.R.)

15 - 63

LES REALISATIONS ET LES PROJETS D'ELECTRICITE DE FRANCE DANS LES
DOMAINE NUCLEAIRE

Roux, J. P.

Revue Générale de l'Electricité, Vol. 67, No. 7, p. 351-64, July 1958.

Realizations and projects of Electricité de France in nuclear field; besides general role of Marcoule and Chinon plants in power generation, paper gives details of electric equipment of latter, particularly of auxiliary services which radically differ from conventional type; description of heat exchanger system between reactor and boilers powered with two motors of 15,000 hp and 3000 rpm.

15 - 64

RECENT PROJECTS ON REACTORS WITH GASEOUS COOLANTS

Kliefoth, W. (Heidenheim)

Atomkern Energie 4, 196-201 (1959) May. (In German).

The significance of the gas cooled nuclear reactor in the programs of the UK, the US and the German Federal Republic are described. Particularly the Advanced Gas Cooled Reactor with CO₂ coolant and the High-Temperature-Gas-Cooled-Reactor with He coolant in Great Britain, and the high temperature reactor of General Dynamics in the US using He as well as a project by Oak Ridge, Kaiser and ACF Industries are discussed. The German project concerns a 15 MWe high temperature reactor developed by the Kernenergiegruppe BBC-Krupp.

15 - 65

REPORT FROM MARCOULE

- - -

Nuclear Power, Vol. 2, No. 17, p. 356-61, September 1957.

Developments in Commissariat à l'Énergie Atomique's industrial center in South of France which includes one of world's first nuclear power facilities, namely Marcoule's reactor G 1 which has thermal output of 40 Mw; features of more powerful reactors designed for producing plutonium and electricity; some particulars on reactor core, gas distribution system, triple pressure cycle, discharge of spent elements, etc; comparison of Marcoule and Calder Hall.

15 - 66 SCOTLAND'S FIRST NUCLEAR POWER STATION

Engineer, Vol. 203, No. 5272, 5273, p. 216-9, February 8, 1957.
p. 260-1, February 15. see also Eng & Boiler House Rev, Vol. 72,
No. 3, p. 85-8, March 1957. Elec Energy, Vol. 1, No. 8, p. 245-50,
April 1957. Nuclear Eng, Vol. 2, No. 11, p. 51-9, February 1957,
6 plates with detailed perspective cutaway diagrams; Constructional
Rev Vol. 30, No. 1, p. 35-6, January 1957.

Station to be built by GEC-Simon-Carves Atomic Energy Group;
plans include two gas cooled graphite moderated natural uran-
ium reactors, each associated with eight heat exchangers, in
which steam is raised at two pressures to drive six 60-mw
turbogenerator sets; between two reactors will be cartridge
cooling pond.

15 - 67 SPECIFICATION OF THE ATOMIC POWER STATION TO BE BUILT BY THE G.E.C.
FOR THE SOUTH OF SCOTLAND ELECTRICITY BOARD

Atomic Energy Rev. 1, 16-17(1957) March.

Not seen.

15 - 68 THE SWEDISH REACTOR
Eklund, S.

J. Nuclear Energy, Vol. 1, No. 2, 93-100 (December 1954).

A new heavy-water reactor commenced operation in Sweden on
July 13, 1954. This paper describes the construction and
gives details of the initial operation of this new reactor.

15 - 69 SYMPOSIUM: CALDER WORKS NUCLEAR POWER PLANT

J. Brit. Nuclear Energy Conf. 2, 41-232(1957) April.

The symposium on Calder Works Nuclear Power Plant, arranged
by the British Nuclear Energy Conference, was held on November
22 and 23, 1956. The nineteen papers cover all the important
features of planning, design, and construction. (auth)

15 - 70 SYMPOSIUM ON CALDER WORKS NUCLEAR POWER PLANT

J. Brit. Nuclear Energy Conf. 2, 233-91(1957) July.

The symposium on Calder Works Nuclear Power Plant, arranged by
the British Nuclear Energy Conference, was held on November 22
and 23, 1956. This issue contains the general reports and the
discussions. (auth)

- 15 - 71 TECHNOLOGIE DES REACTEURS: EXTRACTION DE LA CHALEUR
Erkes, P.
Assn des Ingenieurs Sortis de l'Institut Electrotechnique Montefiore
--Bul, Vol. 68, No. 5, p. 417-33, May 1955.
- Reactor technology: extraction of heat; criteria for choice of coolant include absorption cross section, physical and chemical stability, pumping energy requirement, maximum permissible temperature and type of reactor; comparison of properties of gaseous and liquid coolants.
- 15 - 72 TYPES OF ATOMIC POWER STATIONS EXISTING UNDER CONSTRUCTION AND
PLANNED IN THE USSR
- - -
(AEC-tr-3574) (Translation). 18p.
- An analytical report is made on atomic power plants in the USSR. The information was obtained from Soviet open sources (monographs, periodicals, and newspapers). A description is given of various types of industrial and experimental atomic power stations, and reports on the viewpoints expressed by Soviet scientists as to the stability and efficiency of various reactors are given. Graphic material is included. (A.C.)
- 15 - 73 THE UNITED KINGDOM RESEARCH REACTORS AND THEIR USES
Fenning, F. W. (Atomic Energy Research Establishment, Harwell, Berks, England).
Brit. J. Appl. Phys., Suppl. No. 5, 100-6(1956).
- A brief description and use of the U.K. research reactors is given. Five are in operation: GLEEP, BEPO, DIMPLE, ZEPHYR, and ZEUS. (auth)
- 15 - 74 WORLD'S REACTORS
- - -
Nuclear Eng., Vol. 1, No. 1, 2, 3, 5, 6, 7, 8, 9, p. 14, April 1956.
p. 62, May. p. 106, June. p. 192, August. p. 234, September. p. 276, October. p. 330, November. p. 338, December. (Facing pages).
- Data sheets summarizing principal characteristics and showing cutaway diagrams of reactors as follows: Apr: BEPO--British Experimental Pile; May: CP5--Chicago Pile. June: NRX--Natural Research Experimental. Aug: DIMPLE--Deuterium Moderated Pile. Sept: ZEUS--Zero Energy Uranium System. Oct: Calder Hall Reactor. Nov: 5-MW Power Reactor. Dec: Calder Hall data.

16. GRAPHITE

- 16 - 1 THE ANNEALING OF THE GRAPHITE MODERATOR STRUCTURE IN THE BNL REACTOR
Fox, M. and Powell, R. W.
(TID-10000) Brookhaven National Lab., Upton, N. Y.
January 1954. Decl. March 12, 1957. 17p.

Data are presented from measurements of the magnitude of stored energy contained in reactor graphite, the effect of radiation damage on dimensional stability of graphite, the effects of annealing on graphite growth, and the effect of stored energy release procedures on measurements of physical growth and c-axis growth. Measurements were made on samples of graphite placed in the high flux region of the Brookhaven Reactor in the early days of its history and other samples mined from the graphite structure. Data are presented graphically. (C.H.)

- 16 - 2 THE CALDER HALL GRAPHITE STRUCTURE
Long, E. (UKAEA Industrial Group HQ, Risley, Eng.).
Nuclear Power 3, 58-63(1958) February.

The use of graphite for all structure in the Calder Hall Reactor core presented several novel engineering problems. The design of the graphite structure so as to compensate for thermal expansion, Wigner growth, and neutron leakage is described in some detail. (J.S.R.)

- 16 - 3 CHEMICAL STUDIES OF CARBON DIOXIDE AND GRAPHITE UNDER REACTOR CONDITIONS
Anderson, A. R. and Wright, J. ed. (Atomic Energy Research Establishment, Harwell, Berks, Eng.); Davidson, H. W. (General Electric Co., Wembley, Eng.); Lind, R. (United Kingdom Atomic Energy Authority, Sellafield, Cumb., Eng.); Stranks, D. R. (Univ. of Leeds, Eng.); and Tyzack, C. (United Kingdom Atomic Energy Authority, Culcheth, Lancs, Eng.).
A/CONF.15/P/303. 102p.

The results of five chemical studies of graphite and CO_2 which have received attention in connection with compatibility in Calder Hall Reactors are presented. The thermal reactions between C and CO_2 below 600°C are explained in terms of surface oxide formation. Three successful techniques which have been used to reduce the rate of reaction between graphite and CO_2 at higher temperatures are described. The reaction between

graphite and CO_2 under pile irradiation at temperatures up to 350°C is studied. The results from experiments conducted during startup of the first two Calder Hall Reactors are given. Experiments on the radiation-induced exchange reaction between C^{14}O and CO_2 are described. (W.D.M.)

16 - 4

CONTROL OF RADIATION EFFECTS IN A GRAPHITE REACTOR STRUCTURE
Powell, R. W., Myers, R. A. and Bourdeau, R. G. (Brookhaven National Lab., Upton, N. Y.)
A/CONF.15/P/462. 32p.

A procedure was devised for releasing stored energy under controlled operating conditions. This was accomplished in the Brookhaven Reactor's first annealing operation and was accompanied by a fractional reduction of the dimensional growth which had occurred up to that time. Annealing operations are being conducted at more frequent intervals. As the data of these more frequent annealing operations are studied, perhaps a new and more interesting effect is being observed, namely that the growth is approaching an equilibrium maximum value, which probably depends on the annealing procedures. This view is confirmed by 4 in. long graphite samples that have been irradiated in the pile and measured within ± 0.00003 in. for growth changes after different annealing treatments. These samples show that the percent contraction or recovery of the graphite on annealing is proportional to its growth since the last anneal. This gross growth value is dependent on the short time temperature history of the reactor and not the average temperature. The rate of C-axis increase is approximately ten times the rate of gross growth after a thermal or pile anneal. The changes in dimensions and C-axis as well as the compressive strength and stored energy are discussed. Correlations between some of these properties are presented. It is concluded that sufficient information is now available on the rates of graphite growth under varying conditions and on methods of controlling growth, that graphite structures might be designed with considerably more assurance than in the past. (auth)

16 - 5

DIVISION OF REACTOR DEVELOPMENT PROGRAMS MONTHLY REPORT (FOR) SEPTEMBER 1958.
McEwen, L. H., comp.
(HW-57636 A2) General Electric Co., Hanford Atomic Products Operation, Richland, Wash.
October 15, 1958. 26p. Contract W-31-109-Eng-52.

Gas-graphite reactions were studied for the gas-cooled central station power reactor program.

16 - 6

EVALUATION OF BNL PILE GRAPHITE

Kosiba, W. L., Gurinski, D. H. and Dienes, G. J.
(BNL-255) Brookhaven National Lab., Upton, N. Y.
October 5, 1953. Decl. February 13, 1957. 18p.

The graphite in the Brookhaven Reactor was evaluated with respect to the following properties: stored energy, physical growth, and c-axis change. As a result of these studies, it is believed that: the temperature of the graphite in the BNL pile may be raised to 250°C or even 400°C without an excessive temperature excursion; the low temperature stored energy in the BNL pile graphite may be safely released; some growth recovery should be expected; and it is desirable to anneal the BNL pile. (auth)

16 - 7

GRAPHITE STUDIES FOR THE GAS-COOLED REACTOR PROGRAM

Baker, D. E. General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 185-92.

The rate of oxidation of graphite by carbon dioxide depends upon a number of variables which include the graphite temperature, radiation intensity, gas composition, flow rate, and pressure. Physical properties of the graphite which change the extent of surface area available for reaction also affect the rate of oxidation. Coated graphites, inhibitors or impermeable graphites which would limit oxidation to an acceptable value in an advanced gas cooled reactor are needed. A basic program for the study of the reactions of gases with graphite includes thermal reaction studies and irradiation induced reaction studies. An essential facility in the over-all study of gas-graphite reactions is a recirculating gas loop which will allow proof testing of promising materials and will aid in setting limits on the variables that affect the CO₂-graphite reaction. Although coolant graphite reactions are of primary importance, radiation damage to graphite cannot be neglected. Graphite has been observed to expand when irradiated at temperatures below 300°C and to contract at higher irradiation temperatures. The accumulation of stored energy in the cooler regions of reactors and the decrease in graphite strength as a result of oxidation must also be evaluated. (auth)

16 - 8

PROCEDURE FOR ANNEALING OPERATION

Powell, R. W.

(BNL-1901) Brookhaven National Lab., Upton, N. Y.

October 2, 1953. Decl. February 14, 1957. 11p. Contract (AT-30-2-gen-16).

Information is given covering a procedure for annealing the graphite of the Brookhaven Reactor while in the reactor to alleviate operating problems caused by radiation damage of the graphite. (T.W.S.)

16 - 9

A REVIEW OF GRAPHITE TESTING PROCEDURES AND THEIR APPLICATIONS TO THE BNL REACTOR

Chernick, J. and Kaplan, I.

(BNL-77) Brookhaven National Lab., Upton, N. Y.

November 15, 1950. Decl. December 14, 1955. 32p.

The results of various functional tests of the graphite assigned to the BNL Reactor are collected and reviewed. Methods of improving the precision and accuracy of sigma pile tests are discussed. In particular, a simple method is suggested to account for the nonthermal character of the neutron sources presently in use. The alternative use of a cadmium sheet, interposed between the source and detectors for this purpose, is investigated. It is found that for the usual experimental arrangement, the necessary harmonic corrections depend on both the source spectrum and the position of the cadmium sheet. However, by modifying the experimental arrangement to a spherically symmetrical geometry, these difficulties may be avoided. The conventional solution of the sigma pile equations is inappropriate for a sigma pile of large dimensions. An alternate method is developed for the analysis of experiments on the stacked moderator of a reactor. The neutron flux distribution resulting from a source with a continuous neutron spectrum is considered. The basic case of a point source in an infinite medium is treated. Epithermal and thermal flux distributions are obtained for a Po-Be source on the basis of recent experimental data on the source spectrum. The distributions are compared with those resulting from monoenergetic sources. (auth)

16 - 10

THE STRUCTURE OF THE GRAPHITE MODERATOR AT HUNTERSTON

Gaunt, I. A. B. (General Electric Co., Ltd., London).

G.E.C. Atomic Energy Review 1, 182-7(1958) September.

An account is given of the design of the graphite structure to be used for the reactors of the nuclear power station at Hunterston, in Scotland. A description is given of the method adopted to overcome the effects of Wigner growth, and the means by which the graphite is located and restrained so that free access can be obtained to both ends of the fuel element channels. Information is included on design requirements and the properties of graphite directly affecting design. (auth)

- 16 - 11 SURVEYING GRAPHITE DISPLACEMENTS IN THE WINDSCALE PILES
Powell, D.
(RDB(W)TN-182) United Kingdom Atomic Energy Authority. Industrial Group. Windscale Works, Sellafield, Cumb., England.
Date of MS. November 1954. 8p. (WTSC/R-190).

A simple apparatus was evolved by means of which the vertical distortion produced in the isotope channels due to Wigner growth may be measured. Some results are presented, and recommendations are made for future experiments. (auth)

- 16 - 12 THEORY OF ANNEALING KINETICS APPLIED TO THE RELEASE OF STORED ENERGY FROM IRRADIATED GRAPHITE IN AIR-COOLED REACTORS
Cottrell, A. H. et al. (Atomic Energy Research Establishment, Harwell, Berks, Eng.); and Bell, J. C. and Greenough, G. B. (United Kingdom Atomic Energy Authority, Sellafield, Cumb., Eng.).
A/CONF.15/P/2485. 56p.

The release of stored energy is due to the movement and elimination of lattice defects. A theory is presented of the factors which enable the rate and amount of release to be estimated, for a variety of reactor conditions, in terms of measurements made under a limited set of laboratory conditions. This theory is used to assess various possible methods for operating and annealing graphite moderated reactors so as to avoid both accidental releases of stored energy and unduly high temperatures during deliberate releases. (W.D.M.)

- 16 - 13 VARIATION OF GRAPHITE DAMAGE IN THE REFLECTOR OF BEPO AND THE CALCULATION OF DAMAGE IN GRAPHITE MODERATED PILES
Kinchin, G. H.
(AERE-N/PC-20) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.
October 18, 1951. 11p.

The variation of graphite damage in the reflector of BEPO has been measured by using the change in electrical resistance of graphite. From the results of this experiment it has been possible to calculate the variation of damage with distance from an isolated line of slugs. This provides the data necessary to determine the variation of damage over a cell of any graphite moderated lattice. (auth)

17. HEAT TRANSFER

- 17 - 1 ANALOG REPRESENTATION OF HEAT EXCHANGE. APPLICATION TO THE SIMULATION OF HEAT EXCHANGERS OF NUCLEAR POWER PLANTS
Carteron, J. M. and DeLoux, G. (Electricité de France, Paris, France).
In the Computers In Control Systems Conference, Proceedings. New York, American Institute of Electrical Engineers, 1958. p. 48-53.

This article deals with the simulation of the operation of the heat exchanger of the type useful in Marcoule (G2 and G3) and Chinon (E1). By comparing experimental curves with the simulation diagram and available theoretical curves, very good representation of the actual phenomena can be achieved if the economiser is divided into 12 or 6 cells. This method is compared with J. C. Moise's method and can be generalized to obtain a fairly simple representation of evaporators and of extraction by coolants of the heat generated in the nuclear reactor channels. (F.S.)

- 17 - 2 BASIC PROBLEMS OF HEAT TRANSFER IN NUCLEAR REACTORS
Baehr, H. D.
AEG Mitt. 48, 1-11 (1958) January. (In German)

In nuclear reactors large heat fluxes must be removed at high coolant temperatures, without which the permissible temperatures in the reactor and in the fuel elements would be exceeded. Three problems arising from this factor are considered: heat conduction in the fuel elements, heat transfer to the coolant, and heat removal with the coolant from the reactor. After a comparison of coolants suitable for reactors, the heat transfer problems which occur in gas-cooled, water-cooled, organic liquid-cooled, and liquid metal-cooled reactors are discussed. (tr-auth)

- 17 - 3 A CORRELATION OF HEAT TRANSFER FOR FINNED FUEL ELEMENTS FOR CARBON DIOXIDE REACTORS
Hughes, D. F. and Slack, M. R. (Atomic Power Projects of the English Electric Co. Ltd.; Babcock and Wilcox Ltd; and Taylor Woodrow Ltd., Whetstone, Leics, Eng.).
A/CONF.15/P/38. 24p.

In the course of development of fuel element cans for gas cooled reactors a type of finning has been evolved comprising

multistart helical fins intersected by longitudinal splitters. The flow pattern associated with this type of finning is discussed and the method of testing is described. As the result of tests on a wide range of geometrical arrangements of this finning in different channel sizes it was possible to produce empirical correlations of heat transfer and friction factor. Examples for a series of cans of 2 inches external diameter in a variety of channel sizes are presented. (auth)

- 17 - 4 THE DEVELOPMENT OF FINNED TUBES FOR STEAM-RAISING UNITS
Lymer, A.
G.E.C. Atomic Energy Review 1, 188-94(1958) September.

Two methods are described in which heat-transfer data obtained with atmospheric air flowing across a bank of finned tubes may be transposed for use with hot pressurized carbon dioxide as encountered in steam-raising units. A method to measure the local heat-transfer coefficient over the surface of an annular fin is given, and the use of a water model to give the flow pattern and velocity distribution in a bank of plain or finned tubes is described. The use of a simple wind-tunnel balance to determine the change in pressure-drop coefficient through a bank of plain or finned tubes enables an extrapolation of the pressure drop for any number of rows to be made. (auth)

- 17 - 5 HEAT TRANSFER
Wootton, K. J. (General Electric Co.).
Atomic Energy Rev. 1, 23-6(1957) March.

The heat transfer in gas-cooled reactors is described. The processes involved in the transfer of heat from the uranium to the coolant gas stream and the transfer of heat from the coolant gas to the water and steam are described. (W.L.H.)

- 17 - 6 HEAT TRANSFER EXPERIMENTS FOR HELIUM PLANT
Lyon, R. N.
(CE-501) Chicago. Univ. Metallurgical Lab.
February 25, 1943. Decl. February 14, 1956. 21p. (A-499)

Full scale experiments were conducted to determine the gas film heat transfer coefficients in a proposed helium cooled power plant. The effects of bevelling the edges of the flat plates proposed and of constricting the ducts before and after the element were explored. The results show that the average coefficient for the element as a whole is 3 to 19% more than would be predicted from the ordinarily used Dittus-Boelter equation. However, the coefficient was found to vary locally

within the element in such a way that the film ΔT at the outside of the element is 25 to 50% greater than the average. Beveling the plates and constricting the ducts both reduce the average coefficient, but the former smooths out temperature inequalities in the elements while the latter aggravates them in the ranges explored. Gas velocities up to 735 ft./sec. were used and heat transfer coefficients up to 1300 Btu/hr/ $^{\circ}$ F/ft² and more were found. (auth)

17 - 7

HEAT TRANSFER IN A CROSS FLOW NUCLEAR REACTOR
Larson, C. L. (Raytheon Mfg., Co., Waltham, Mass.).
Nuclear Sci. and Eng. 4, 607-22(1958) November.

The heat transfer between mutually perpendicular fuel and coolant channels of a nuclear reactor is studied for both isotropic and anisotropic conducting media. By means of an electrical conduction analog, a characteristic length of a unit cell of the reactor is experimentally determined for channels of rectangular and circular cross section. Empirical expressions for this characteristic length are given which closely fit the experimental data. A finite difference calculation is used to determine the temperature distribution within a unit cell. (auth)

17 - 8

HEAT TRANSFER IN AN ANNULUS WITH ASYMMETRIC HEATING
Lisin, A. V., Mackewicz, W. V. and Reynolds, W. C.
(IDO-28534) Aerojet - General Nucleonics, San Ramon, Calif.
February 28, 1959. 40p. "ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM".

The objectives of this investigation include: determination of the validity of using tube flow correlations for calculating pressure drop and heat transfer characteristics in thin annuli; investigation of the validity of using Stein's heat flux asymmetry correction for parallel planes to calculate the heat transfer in asymmetrically heated thin annuli; and investigation of the effects of eccentricity on pressure drop, heat transfer, and temperature distribution in concentric ring fuel elements. (auth)

17 - 9

HEAT TRANSFER USING CALDER HALL TRANSVERSE FINS
- - -

Power Reactor Technology: Technical Progress Reviews, Vol. 1, No. 1,
December 1957. p. 5-10.

Not looked at.

17 - 10

HEAT TRANSFER WITHIN REACTOR FUEL ELEMENTS

Skipper, R. G. S.

G.E.C. Atomic Energy Rev. 1, 141-6 (1958) March.

A typical fuel element for a gas-cooled nuclear reactor consists of a cylindrical rod of uranium sealed into a can of magnesium alloy. Heat produced in the uranium by fission is removed by gas flowing over the can, and in consequence there is a temperature drop between the center of the rod and the inside of the can. This consists of two parts: the drop due to heat conduction in the uranium; and the drop due to contact resistance between the fuel and the can. The method of calculating the first of these is outlined, and experiments to measure the second are described. (auth)

17 - 11

HIGH-TEMPERATURE SYSTEMS FOR NUCLEAR PROCESS HEAT

McGee, J. P. Bureau of Mines, Morgantown, W. Va.

In Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 305-24.

Investigations are being conducted to determine the feasibility of using nuclear heat in chemical processing. The use of nuclear reactors to furnish high-temperature heat is a field that has not received as much attention as electric power production and propulsion. As part of the Atomic Energy Commission--Bureau of Mines process heat reactor program, a system is being studied in which helium is used as a heat-transfer medium from the nuclear reactor to the chemical processing vessel. A vessel heated by electrical induction to simulate a nuclear fission source of heat has been designed and constructed and is undergoing preliminary testing. Gasification of coal and re-forming of hydrocarbons are promising uses for nuclear process heat. (auth)

17 - 12

INTENSIFICATION OF HEAT TRANSFER IN A GAS-COOLED NUCLEAR REACTOR

Berger, F. and Stach, V. (Czechoslovakia).

A/CONF.15/P/2486. 17p.

The possibility of utilizing the partial ionization of the coolant gas in a reactor to increase the heat removal rate from fuel elements is investigated. Results from an experiment conducted in a gas-cooled loop are reported. It is concluded that in a gas-cooled reactor a considerable increase of heat transfer from the fuel element may be achieved by applying an electric field. The magnitude of the increase depends on the density of gas ionization, applied voltage, and the flow rate of the gas. (W.D.M.)

17 - 13 STRESS RELIEVING ON THE CALDER HEAT EXCHANGERS

(NP-6633) United Kingdom Atomic Energy Authority. Industrial Group
H. Q., Risley, Lancs, England. Report No. 5256.
September 8, 1955. 32p.

A detailed account of the high-frequency induction heating equipment and its method of operation at Calder Hall is reported. Drawings, photographs, and wiring diagrams are given. (auth)

17 - 14 SUPPLEMENTARY ENGINEERING TESTS FOR THE BROOKHAVEN REACTOR

Bareis, D. W. and Page, W. R.
(BNL-46) Brookhaven National Lab., Upton, N. Y.
November 15, 1950. Decl. February 14, 1957. 114p.

An extensive thermal analysis has been made of the structure of the BNL reactor. Use was made of the electric analog technique to obtain detailed temperature patterns for various parts of the reactor structure for a number of different operating conditions. Structural changes recommended on the basis of the temperature information thus obtained were incorporated into the analogs to test the effectiveness of the changes. The analog data for both the original and final designs are presented along with a description of two auxiliary engineering tests which were run to determine air flow patterns over portions of the reactor structure where the amount of forced convection heat transfer would otherwise have been an uncertain factor. Detailed calculations of surface heat transfer coefficients simulated in the analogs are presented in appendices. (auth)

17 - 15 SYSTEMATIC INVESTIGATION OF HEAT TRANSFER AND FLOW RESISTANCE OF FIN TUBES, A CONTRIBUTION TO THE USE OF SPECIALTY TUBES BY GAS-COOLED REACTORS

Grass, G. and Coenen, F. P.
Atomkernenergie, 4, 41-48, (1959) February (In German)

The present investigations refer to measurements of heat transfer and pressure drop of single brass fin tubes for cross flow. The influence of fin height, fin spacing and Reynolds number on both qualities of tubes is graphically plotted. With certain restrictions the results on single brass tubes can be transferred to banks of steel tubes, because all measurements on fin tubes are plotted according to the relationship of their qualities to the plain tube. The volume as well as the pressure drop of fin tube heat exchangers - plotted against fin height with constant total heat input - pass a minimum at fin heights between 5 and 10 mm.

THERMAL EXCHANGES IN GAS REACTORS

Le Foll, J., Gelin, P. and De Robert, E. (Commissariat à l'Énergie Atomique, Paris).
A/CONF.15/P/1145. 51p.

Both convection and conduction come into play simultaneously in the transfer of the heat from the fuel to the gas, and these two phenomena are quite different, both as to their nature and as to the present knowledge of the laws which apply to them. An endeavor is made to distinguish between these two phenomena to the fullest possible extent in order to restrict the experimentation to an investigation of forced convection and to the indispensable over-all verifications. (W.D.M.)

18. INSTRUMENTATION

- 18 - 1 BROOKHAVEN REACTOR GAMMA FLUX BY CALORIMETRY
Carroll, J. G. and Bolt, R. O.
(AECU-3933) California Research Corp., Richmond, Calif.
October 22, 1958. 38p. Contract AT(11-1)-174.

Simple calorimeters were built comprising an insulated lead rod so constructed that heat flowed in one direction. Heat induced in the lead by gamma rays caused heat to flow, producing a temperature gradient which was used as an index of gamma dose rate. The calorimeters were calibrated using accepted standards in the Canal Gamma Source at the Materials Testing Reactor. Calorimeters were then exposed in Hole E-25 for the Brookhaven Graphite Reactor while the reactor was operating at 25 megawatts on natural uranium. Gamma flux at the reactor center line was found to be 10^{12} photons/cm²/sec. (auth)

- 18 - 2 LE CONTROLE HYGROMETRIQUE DES GAS DE REFROIDISSEMENT D'UN
REACTEUR. (Hygrometric Control of Reactor Cooling Gas.)
Briot, A. and Turret, J. (Campagne des Compteurs, Montrouge,
France).
A/CONF.15/P/1151. 23p.

In gas-cooled reactors, the humidity may cause the corrosion of fuel cans, especially if they have a magnesium base. A hygrometric control installation must consist of a measurement station, alarm system, and a repair station. The factors which determine the choice of a hygrometer are discussed. The operation principles and the performance limitations of a dew-point hygrometer are described in detail. Descriptions are given of the measurement posts and of the alarm stations. The installation of each is discussed. A method for locating the deficient element in the heat exchanger is given. (J.S.R.)

- 18 - 3 DETECTION OF BREAKS IN THE CLADDING IN GAS PHASE COOLED REACTORS
Labeyrie, J. and Roguin, A.
In International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Proceedings. Vol. 3, Power Reactors. New York, United Nations, 1956. p.86-90.

In a nuclear reactor using rods or plates cooled by a gaseous fluid, breaks or cracks may appear in the cladding which provides gas-tight separation between the coolant and fissionable material. Radioactive fission products may escape through these breaks and thus pollute the outlet side. Also, the gas which penetrates through the break may react with the metal and cause the slugs or plates to swell. A detection device designed to keep a check on all the fuel channels in a reactor is described. The operation and construction of such a device are indicated for the case of a reactor using natural uranium cooled by air circulation at atmospheric pressure.

18 - 4

A FISSION-PRODUCT DETECTION SYSTEM

Mills, J. B. B. and Hartshorn, P. F. (General Electric Co., Erith, Eng.) and Bryden, R. D. (Salford Electrical Instruments, Ltd.) Atomic Energy Rev. 1, 97-104(1957) September.

A description is given of a fission-product detection system for gas-cooled reactors. A schematic diagram is given for a fuel-channel sampling system. (W.L.H.)

18 - 5

INSTRUMENTATION, 1958

- - -

Nuclear Power 3, 148-86 (1958) April.

An over-all picture of the instrumentation requirements of modern nuclear power and research reactor installations and a review of the latest British instruments available to meet them are given. Five review articles by the country's leading authorities are included. These cover physical or non-nuclear instruments needed in a power station, lines of development for instrumentation, methods of neutron detection by means of gas ionization devices, scintillation counting methods, and methods of finding neutron flux variations. Diagrams are included showing the main instrumentation features in a gas-cooled power station, a high-flux reactor, and a fast breeder reactor. Preview features of two exhibitions in London--The Physical Society and the Instruments, Electronics, and Automation exhibitions--are included. (M.H.R.)

18 - 6

LEAK DETECTION IN NUCLEAR ENERGY HEAT EXCHANGERS

Crumley, P. H.

Nuclear Energy Engr. 12, 448-9(1958) December.

A sensitive CO₂ leak detector for heat exchangers of gas-cooled nuclear reactors is described. The equipment operates in the following manner: Steam from the power plant is continuously sampled and any traces of CO₂ which may be carried by the steam are separated in the de-gasser and subsequently measured by an infra-red CO₂ analyser. A significant increase in the CO₂ content of the steam would indicate a leak between the CO₂ circuit and the steam circuit of the heat exchanger. (J.H.M.)

- 18 - 7 MONITOR FOR WATER VAPOUR IN CARBON DIOXIDE
Reid, C. D., Paget, K. W. and Hill, J. W.
(IGR-R/CA-201)(D) United Kingdom Atomic Energy Authority.
Industrial Group. Capenhurst Works, Capenhurst, Ches., England.
December 1957. 17p.

A review is made of the duties of a hygrometer working as a water-vapor monitor for the CO₂ coolant of the Calder Hall type of reactor. Of the types of hygrometer considered for this application, only the Infra-Red Gas Analyzer proved suitable. The performance of the Infra-Red Gas Analyzer as manufactured can be improved by a slight modification, details of which are given. This modified instrument is considered to be the more acceptable for the plant application. (auth)

- 18 - 8 NUCLEAR POWER PLANT INSTRUMENTATION

Brit Communications & Electronics, Vol. 4, No. 1, p. 26-30.
January 1957.

Control and instrumentation features of power reactors at Calder Hall plant in England; how instruments are disposed in number of control centers; means for control of reactor heat production and utilization; detection of faulty fuel elements; role of electronics in instrumentation.

- 18 - 9 POWER STATION INSTRUMENTATION
Nutter, J. C. and Dowsing, J. (Elliott Brothers, Ltd., London).
Nuclear Power 2, 455-62(1957) November.

The design of instrumentation for a gas-cooled plant is given. Schematic diagrams are presented of major reactor and steam generator instrumentation. (W.L.H.)

18 - 10

SOME NON-NUCLEAR MEASUREMENTS ON THE CALDER HALL PLANT

Sandiford, K. R.

In Fifth Tripartite Instrumentation Conference, Brookhaven National Laboratory, October 22-26, 1956, Minutes. (TID-7543), February 1958.
p. 24-5.

Rate of change of pressure of CO_2 system, unbalance of coolant flow, dissolved oxygen in the feed water, leakage of CO_2 coolant.

19. LOADING AND UNLOADING

- 19 - 1 FUEL ELEMENT LOADING APPARATUS FOR NEUTRONIC REACTORS
Anderson, G. A. (to U. S. Atomic Energy Commission).
U. S. Patent 2,794,562. June 4, 1957.

A device for remotely refueling a gas cooled reactor without loss of the high pressure gas is described. A rotating magazine containing the fuel elements is positioned against the core of the reactor, rotated to align the cartridges with the desired tubes, and a remotely controlled plunger arrangement inserts the fuel elements into the reactor channels. This charging apparatus simplifies the refueling of a gas cooled reactor. (auth)

- 19 - 2 IMPROVEMENTS IN OR RELATING TO FUEL ELEMENT CHARGING/DISCHARGING
GEAR FOR NUCLEAR REACTORS
Long, E. (to United Kingdom Atomic Energy Authority).
British Patent 772,702. April 17, 1957. Nuclear Power 2,
394(1957) September.

In the Calder Hall reactors, to reduce the number of standpipes through which fuel is loaded into the 1,696 core channels, the latter are grouped in 16's with a single charge tube serving each group. Since the elements thus approach the channels obliquely, means have to be provided to guide them in without damage to the graphite core or the element. The devices used at Calder to do this are known as charge guides or pans and this is the patent covering them. The guide is a nodular iron casting with a 4 x 4 array of tapered holes corresponding with a fuel channel group. The holes are 11 in. long and taper from 5 in. to 5.92 in. diam. Support of the guides on the top of the graphite core is at three points since thermal movements of the blocks are to be expected during the reactors' lifetime. Location is centered on the 17th channel at the center of the guide which serves the control rod hole and which also receives the snout of the charging machine. The tapered holes of the charge pan are each fitted with an entry for the burst slug detection gear 'sniffer' pipes. It is stated that these guides weigh 600 lb. (auth)

- 19 - 3 LOADING AND UNLOADING
Ertaud, A. and Derome, G.
Bull. inform sci. et tech. No. 20, 69-88(1958) August (In French)

The loading and unloading system of the G-2 and G-3 reactor consists essentially of a mobile loading device, a lattice of inclined drains and pipes which collects the irradiated fuel elements, and telecommanded workshops which group the fuel elements and transport them to the pool. Each of these parts is considered in detail. (J.S.R.)

19 - 4 LES PROBLEMES DU CHARGEMENT ET DECHARGEMENT DES ELEMENTS COMBUSTIBLES
DANS LE REACTEUR E.D.F.1 (Loading and Unloading Problems for the
Reactor E.D.F.1).

Leo, Kaplan, and Segard (Electricite' de France, Paris).
A/CONF.15/P/1201. 25p.

The EDF-1 reactor has two loading and unloading devices of completely different design. The main device, which operates when the reactor is shut down and the pressure vessel is under pressure, is the lower unloading system and it has a single arm which serves all the channels. The description of the lower system, its operation, and the details of its use are given. The upper unloading device, designed to operate when the reactor is shut down and the pressure vessel is without pressure, is used when the lower device fails. The loading and unloading is performed through the control rod holes. A detailed description is given of the upper device and its operation. (J.S.R.)

19 - 5 REACTOR UNLOADING
Leverett, M. C. (to U. S. Atomic Energy Commission).
U. S. Patent 2,824,056. February 18, 1958.

This patent is related to gas cooled reactors wherein the fuel elements are disposed in vertical channels extending through the reactor core, the cooling gas passing through the channels from the bottom to the top of the core. The invention is a means for unloading the fuel elements from the core and comprises dump valves in the form of flat cars mounted on wheels at the bottom of the core structure which support vertical stacks of fuel elements. When the flat cars are moved, either manually or automatically, for normal unloading purposes, or due to a rapid rise in the reproduction ratio within the core, the fuel elements are permitted to fall by gravity out of the core structure thereby reducing the reproduction ratio or stopping the reaction as desired.

RETAINING DEVICES FOR FUEL ELEMENTS IN GAS-COOLED NUCLEAR
REACTORS

Robinson, G. E. (to United Kingdom Atomic Energy Authority).
British Patent 794,891. Nuclear Eng. 3, 556(1958) December.

Gas pressure may expel fuel elements from the horizontal coolant channels. Then a new string of elements has to be introduced and the discharged elements have to be reprocessed. In order to prevent this happening, retaining devices are attached by a coupling at one end of the coupled string of fuel elements and hooks at the other end engaging the face of the reactor core. The hooks are in the form of pivoted cranks with a locking member to lock them in position. These members can be unlocked to release the cranks from their engagement with their reactor core. (auth)

20. MAINTENANCE

20 - 1 INSPECTION AND MAINTENANCE OF LARGE REACTORS AT WINDSCALE WORKS AND CALDER HALL

Tuohy, T. and Ireland, G. D. (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.).
A/CONF.15/P/1518. 44p.

An account is given of the maintenance and remedial work which has been done within the biological shield of the Windscale reactors, and covers: (a) Retention of reactor fuel elements which were tending to blow out of the graphite channels. (b) Remote welding and cutting operations using oxy-acetylene equipment at distances of up to 20 feet from the operating position. (c) Repair work on the fuel element detection gear which involved the entry of personnel into the biological shield. Details are given of the health physics control and of radiation exposures received by personnel during the maintenance of the fuel element detection gear. No maintenance work has been necessary within the biological shields on Calder Hall Reactors, but both heat exchangers and gas inlet ducts have been entered for inspection. The result of such inspections and health physics data are given. (auth)

20 - 2 MAINTENANCE OF VARIOUS REACTOR TYPES

Draper, B. D.
(CF-57-4-92) Oak Ridge National Lab., Tenn.
April 8, 1957. 26p. Contract (W-7405-eng-26).

Maintenance of reactor power plants is more complex than conventional fossil fired steam turbo-generator plants because of the presence of radioactivity either localized in the reactor vessel or scattered throughout the circulating loops depending upon the reactor type. It is obvious that the easiest systems to operate and maintain are those which permit the use of conventional tools and techniques on a large portion of the reactor auxiliaries. Following is a list of reactor power plant types, arranged in order of increasing difficulty of maintenance in the opinion of the author with advantages and disadvantages maintenance-wise of each (auth)

21. MATERIALS

21 - 1 ADVANCED MATERIALS DEVELOPMENT

Coobs, J. H. Oak Ridge National Lab., Tenn.

In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 363-72.

The materials problems associated with the design of advanced gas-cooled reactor systems are discussed, and a number of materials combinations are evaluated. Preliminary results of current development work on fuel elements and capsule materials are presented. Three of the most favorable materials combinations are selected and proposed for further study.
(auth)

21 - 2 APPLICATION OF MAGNESIUM TO REACTOR DESIGN

Untermeyer, S. Argonne National Lab., Lemong, Ill.

In Nuclear Science and Technology, (TID-2503), (Del), 1958.
(ANL-WPB-335). p. 443-6.

The properties of magnesium are reviewed and its uses in fuel elements and in the structure of air-cooled reactors are discussed. (T.R.H.)

21 - 3 BORON STEEL FOR BROOKHAVEN NATIONAL LABORATORY

Kemmer, F. R.

(HKF-3) Ferguson (H.K.) Co., Inc., New York

April 1, 1949. Decl. September 21, 1956. 51p. For Brookhaven National Lab. Contract AT-30-2-GEN-16, (Subcontract) A-E-M- S-26.

The pile at Brookhaven National Laboratory will be controlled through the operation of sixteen boron steel bars which will be inserted horizontally into the pile. Each of the control rods will be operated through a toothed rack mechanism to which each control rod is attached. The end section of the toothed rack will be spliced to one end of the boron steel rod. The boron steel control rods for the Brookhaven pile are much longer than any boron steel control rod made in the past. The boron content is higher than used heretofore and an unexpected new requirement developed with respect to manganese content. A maximum content of 0.10% manganese was specified but the aim was preferably to make heats of steel containing boron with practically zero manganese content. This new requirement on manganese resulted from the fact that at least one of the manganese isotopes is radioactive. Chemical specifications

and heat analysis data are presented for boron steel for use in these boron steel control rods.

21 - 4 CREEP TESTS ON MATERIALS FOR GAS-COOLED REACTORS. NEW RESEARCH EQUIPMENT AT G.E.C. ATOMIC LABORATORIES

- - -

Metallurgia 56, 313-14(1957) December.

Research equipment designed to provide creep and stress-rupture data on metals and alloys in gaseous atmospheres at high temperatures and pressures has been installed. Valuable basic work has been done in investigating the properties of magnesium alloys, widely employed for fuel element cans, in carbon dioxide atmospheres. The design of the equipment is such that it is comparatively easy to adapt it for operation in other atmospheres. The machines now in use provide an efficient means of obtaining very rapidly a broad assessment of the suitability of materials for reactor service. (A.C.)

21 - 5 MAGNESIUM CANNING

- - -

Metal Industry, Vol. 94, No. 16, p. 306, 309, April 17, 1959.

Magnesium alloy, Magnox Al2, used as canning material for Calder Hall reactors and proposed for use in G.E.G.B. and S.S.E.B. reactors; reasons for its selection; properties required in canning materials for this type of reactor properties of Al2 examined against those of possible alternative magnesium alloys.

21 - 6 MATERIALS COMPATIBILITY AND FUEL ELEMENT TEST PROGRAM

Trauger, D. B. Oak Ridge National Lab., Tenn.

In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 165-84.

Metals and alloys of interest as possible materials of construction for a gas-cooled reactor and reactor-grade graphite were heated together in vessels containing static helium or in helium filled natural convection and forced convection loops. From these tests, measures of the inertness of various test specimens to the gaseous impurities evolved from graphite on heating were obtained. The proposed fuel element test program is given. The variables under investigation are discussed. (W.D.M.)

- 21 - 7 METALLURGICAL ASPECTS OF CALDER HALL
Bishop, T. (John Miles & Partners., London).
Metal Progress 71, No. 6, 65-71(1957) June.

Some of the metallurgical problems involved in the construction of the Calder Hall Reactors are discussed. Among these are U fabrication and canning, selection of steels for pressure vessels, and pressure vessel erection and stress relief. (W.L.H.)

- 21 - 8 PIPING INSTALLATION FOR BURST-CARTRIDGE DETECTION GEAR IN
CALDER HALL REACTORS
Laithwaite, J. M.
Brit Welding J, Vol. 4, No. 8, p. 360-7, August 1957.

Design, joining techniques, preparatory work, and installation of 45 mi of stainless steel piping described; steps in developing brazing method; of nearly 12,000 copper brazed joints made with high frequency induction heating equipment, only 111 failed to pass initial leak test.

22. OPERATION

22 - 1 BROOKHAVEN NATIONAL LABORATORY PILE OPERATING MANUAL

--
(M-4414) Brookhaven National Lab.

Book No. 2. April 1949. Decl. January 13, 1956. 161p. Contract (AT-30-2-gen-16).

Included are operating procedures for the BNL Reactor console, fan house, cooling tower, He system, reactor loading and unloading, experimental facilities, and the filters and stack. (B.J.H.)

22 - 2 CALDER HALL COMPLETES FIRST TWO YEARS

Sykes, J. H. M.

Elec World, Vol. 151, No. 13, p. 67-70, March 30, 1959.

Operating experience with plant showing flexibility and basic reliability of installation; problems encountered with fuel elements; distribution of neutron flux; relation of reactivity and temperature; reduction of frequency of Wigner energy release; reactor shutdown procedure.

22 - 3 COMMISSIONING AND OPERATION OF 'A' STATION, CALDER WORKS

Davey, H. G. (U.K.A.E.A., Industrial Group, Windscale Works, Cumberland, Eng.)

J. Brit. Nuclear Energy Conf. 3, 101-8(1958) April.

The commissioning and operation of "A" station, Calder Works consisting of reactors one and two are discussed. Just the reactor and its associated equipment are considered. Particulars are given of the loading of fuel, the measurement of lattice constants, and the installation and calibration of control rods. A brief description of plant tests and the work done to make the plant operative is given. (W.D.M.)

22 - 4 THE COMMISSIONING AND OPERATIONAL EXPERIENCE OF THE CALDER HALL REACTORS

Marsham, T. N.

(IGT-R-20) United Kingdom Atomic Energy Authority. Industrial Group H. Q., Risley, Lancs, England.

October 1957. 14p.

The initial loading, testing, and operation of the Calder Hall reactors are reviewed and test results are compared with calculated data. (D.E.B.O)

- 22 - 5 DESCRIPTION AND OPERATION OF TWO ORNL RESEARCH REACTORS --X-10 GRAPHITE REACTOR AND LITR
Cox, J. A. (Oak Ridge National Lab., Tenn.).
Am. Ceram. Soc. Bull. 36, 90-5(1957) March.

Two types of research reactors are operated at Oak Ridge National Laboratory. One is a graphite moderated, air-cooled, natural uranium reactor; the second is a water-cooled and moderated, enriched uranium reactor. The advantages and characteristics of each type are discussed and some of the problems encountered in their operation are described. (auth)

- 22 - 6 EXPERIENCE WITH B.E.P.O.
Jackson, R. F.
In International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955. Proceedings. Vol. 2, Physics: Research Reactors. New York, United Nations, 1956. p. 266-269.

Discusses the development of experimental usage and history of operation from initial divergence in July, 1948 to 1955. B.E.P.O. provides the bulk of the radioactive isotope production of the United Kingdom. It has provided a very large volume of experimental flux coupled with extreme simplicity of use and lack of stringent safety requirements in respect of design and compatibility of experiments.

- 22 - 7 OPERATING EXPERIENCE AT CALDER HALL
Davey, H. G., Gawthrop, J. and Marsham, T. N. (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.).
A/CONF.15/P/1522. 17p.

An account is given of the operating experience with the Calder Hall nuclear power plant. The operation of the control system and the degree of operational flexibility available are described together with the various engineering and nuclear limiting features. The performance and adequacy of important plant items such as the instrumentation and safety circuits, fuel elements, fuel handling equipment, and burst cartridge detection gear are discussed. Radiation levels experienced in operation and during maintenance work are referred to. It is concluded from the experience of 18 months operation that the Calder Hall reactors are satisfactory in service and possess a degree of reliability and flexibility which indicate that the gas cooled reactor system is well suited to the task of electrical power generation. (auth)

- 22 - 8 OPERATION OF THE ORNL GRAPHITE REACTOR AND THE LOW-INTENSITY TEST REACTOR--1954
Rupp, A. F. and Cox, J. A.
(ORNL-1866) Oak Ridge National Lab., Tenn.
September 16, 1955. Decl. August 14, 1956. 23p. Contract W-7405-Eng-26.

Operations data for 1954 on the Graphite Reactor and LITR are reported. Available reactor data of general interest are appended. Graphite reactor information includes operating temperatures, shielding, control system, cooling system, and a new water demineralizer system. LITR information includes data on lattice, shielding, control system, flux measurements, and the cooling system. (D.E.B.)

- 22 - 9 OPERATION OF THE ORNL GRAPHITE REACTOR AND THE LOW-INTENSITY TEST REACTOR, 1955; LITR FLUX TRAVERSES
Rupp, A. F. and Cox, J. A. (comps.). (Compiled from reports by: Casto, W. M., Cagle, C. D., Tabor, W. H. and Weeks, T. C.).
(ORNL-2071) Oak Ridge National Lab., Tenn.
August 10, 1956. Decl. March 2, 1957. 40p. Contract W-7405-eng-26.

The ORNL Graphite Reactor operated very well during 1955. The downtime was low, only 8.6%. The fuel in the bonded slugs did not perform as well in 1955 as in 1954. Much of the trouble was undoubtedly due to growth of slugs which were not beta-transformed. It is known that some slugs had grown over 1/2 in. The automatic control system installed in 1954 continued to operate satisfactorily. The cooling system gave minor trouble when one of the 900-hp fan motors had to be replaced because of shorts in the rotor. The high radiation in the canal was the largest source of trouble. Approximately 55 tons of slugs discharged from the reactor in 1952 were sent to the Metal Recovery Plant. Enough slugs had ruptured, due to their long exposure in the canal and reactor, to contaminate badly of water. Most of the contamination was removed by the end of the year, but the radioactivity which had soaked into the canal walls was enough to give high radiation fields. A solution to this problem was being sought at the end of the year. A study is under way on the possibility of increasing the flux of the ORNL Graphite Reactor by reloading with enriched uranium fuel. Results of the study appear to be promising. The LITR operated very satisfactorily in 1955. No major troubles developed, and the usage for research increased steadily. More facilities were installed for radioisotope production, and at the end of the year a project was under way to make a smaller grid assembly, thus exposing more of the lattice for experiments. Another project was under way to provide a third vertical facility which will not have to be removed each time the top plug is taken off. The cycle time for major shutdowns was increased to three weeks,

which will cause less interference with research and operations. The downtime was high because of over 1000 hr of scheduled and 150 hr of unscheduled shutdowns for research. The utility and versatility of the LITR were well demonstrated by the 149 special scheduled shutdowns for research which enabled experimenters to obtain important data. A test of remote control of the LITR from the Graphite Reactor control room was begun in October and was still under way at the end of the year. The results appear very promising to date. A large number of scrams, apparently originating in the period amplifier or period system, was one of the worst troubles encountered. However, no single cause for the spurious scrams could be found, and they continued to occur at varying frequencies throughout the year. Improvements continued in the demineralizer system, and the amount of N_p^{239} in the water decreased by a factor of about 6. This appears to indicate that the N_p^{239} was being formed from natural uranium contamination in the reactor and that the source has been partly removed. (auth)

22 - 10

OPERATION OF THE ORNL GRAPHITE REACTOR AND OF THE LOW-INTENSITY TEST REACTOR AND PREOPERATIONAL WORK ON THE ORR

Cox, J. A. and Casto, W. R. Annual Report for 1957. Compiled from reports by Cagle, C. D., Gaither, C. B., Tabor, W. H. and Weeks, T. C.

(ORNL-2562) Oak Ridge National Lab., Tenn.

December 9, 1958. 29p. Contract W-7405-eng-26.

The ORR Building and associated facilities were turned over to the Laboratory July 15, 1957. The first six months of the year were spent in designing major additions and equipment, in fabricating and procuring equipment, and in preparing for testing and operation. A mockup of the reactor mechanical controls was assembled and then tested under conditions simulating operation conditions as closely as possible. Further activities included training of personnel, performing hydraulic tests on the reactor, and preparing for the neutron tests. (A.C.)

22 - 11

OPERATIONAL EXPERIENCE AT CALDER HALL

Stretch, K. L.

J. Brit. Nuclear Energy Conf. 3, 109-15(1958) April. Also Instn. Elec. Engrs. Proc., Vol. 105, pt. A, No. 23, October 1958. p. 475-83.

The program carried out to commission and operate the Calder Hall nuclear power station from the time when reactor No. 1 was taken over by the operations branch

in May 1956, until the complete unit was on line at full power is described. The relevance of the tests and their results to the successful operation of the Calder Hall project, in particular and the prospects for nuclear generation of power in general, is considered. (W.D.M.)

22 - 12

RESEARCH PROGRAM AND OPERATING EXPERIENCE ON ORNL REACTORS

Ramsey, M. E. and Cagle, C. D.

In International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Proceedings. Vol. 2, Physics: Research Reactors. New York, United Nations, 1956. p. 281-294.

Description of the ORNL graphite-moderated, air-cooled, normal uranium reactor, its operation, and the research facilities available. The description includes the core, the shield, the coolant system, and the control systems. Fuel element failures are noted and various fuel loading patterns are illustrated. Research facilities include openings for instruments; for exposing samples at different fluxes, times, temperatures; and tunnels for exposure of various biological specimens.

23. PEBBLE BED DESIGN

23 - 1 DESIGN AND FEASIBILITY STUDY OF A PEBBLE BED REACTOR-STEAM POWER PLANT

(NYO-8753) (Vol. 1) Sanderson and Porter, New York and Alcoa Products, Inc., (Schenectady, New York.)
May 1, 1958. 290p. Contract AT(30-1)-2095.

Originally issued as S and P 1963A, Parts I and II. A design and feasibility study of a pebble bed reactor-steam power plant is presented. The reactor design which evolved from this study is a 125 Mwe helium-cooled two-region thermal breeder, operating on the uranium-thorium cycle, in which all core structural materials are graphite. Fuel is in the form of unclad spherical elements of graphite, containing fissile and fertile material. The primary loop consists of the reactor plus three steam generators and blowers in parallel. Nuclear characteristics, costs, etc., are given. (W.D.M.)

23 - 2 GAS COOLED PEBBLE BED REACTOR FOR A LARGE CENTRAL STATION. REACTOR DESIGN AND FEASIBILITY STUDY

Schock, A., Bruley, D. F., Culver, H. N., Ianni, P. W., Kaufman, W. F., Schmidt, R. A. and Supp, R. E.
(CF-57-8-12) Oak Ridge School of Reactor Technology, Tenn.
August 1957. 323p.

An optimum economic design for a high temperature, helium cooled, central station reactor power plant of about 400 Mw of electric power was determined. The core consists of a randomly packed bed of unclad graphite spheres, approximately one in. in diameter, impregnated with U^{233} and thorium such that a conversion ratio of near unity is achieved. The high temperature helium permits steam conditions, at the turbine throttle, of 1000°F. and 1450 psia.
(auth)

23 - 3 GAS COOLED PEBBLE FILES

Amorosi, A.
(M-4079) Clinton Labs., Oak Ridge, Tenn.
August 7, 1947. Decl. December 5, 1955. 6p.

A gas-cooled cylindrical pebble pile in which the gas flows axially from one face of the cylinder to the other will produce in the order of 1/2 the specific power

(kw output per unit pile void volume) of a similar size pile in which the flow is in continuous channels for the same operating pressure, temperatures, etc. This is evident from the fact that the heat transfer rate per unit friction power is many times poorer (about 20 times poorer) for a pebble bed than for flow inside of tubes. A quantitative comparison of the size of a pebble pile vs. a continuous channel pile is tabulated. (auth)

- 23 - 4 A HIGH TEMPERATURE-GAS CYCLE PEBBLE BED REACTOR FOR CENTRAL STATION USE
Robinson, S. T. and Benenati, R. F., Sanderson and Porter, New York. In Information Meeting on Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p.193-215.

A high-temperature gas cycle Pebble Bed Reactor is described together with the specific design and economics of a 125 Mwe central station power plant. The nuclear characteristics of the system are presented together with a discussion of fuel cycles, fuel element types, and development problems. (auth)

- 23 - 5 HIGH-TEMPERATURE PEBBLE-BED REACTOR--STEAM-POWER PLANT OF 125 ELECTRICAL Mw
Benenati, R. F.
AIEE--Trans, Vol. 77, pt 3 (Power Apparatus & Systems), No. 40, p. 1329-34, February 1959.

Design of 2-region thermal breeder reactor, operating on uranium-thorium cycle, in which all core structural materials are graphite, with helium as coolant fluid; helium which circulates in primary loop of power plant transports heat of fission from reactor to steam generator; steam is used in secondary loop to generate electricity in conventional turbine generating equipment. Paper 58-1110.

- 23 - 6 NEUTRONIC REACTOR SYSTEM
Daniels, F. (to U.S. Atomic Energy Commission).
U. S. Patent 2,809,931. October 15, 1957.

Gas-cooled solid-moderator type reactors wherein the fissionable fuel and moderator materials are each in the form of solid pebbles, or discrete particles, and are substantially homogeneously mixed in the proper proportion and placed within the core of the reactor are described. The shape of these discrete particles must be such that voids are present between them when mixed through the voids between the fuel and moderator particles to

absorb the heat generated by the chain reaction. The hot helium gas is drawn off the top of the core and may be passed through a heat exchanger to produce steam.

- 23 - 7 THE PEBBLE-BED HIGH-TEMPERATURE REACTOR FOR 15 MW (ELECTR.)
Schulten, R. (Brown, Boveri & Cie. AG. and Fried. Krupp, Ger.).
A/CONF.15/P/1054. 15p.

The problems concerning the construction of a gas-cooled high-temperature reactor are presented. The behavior of the reactor within the temperature range of 600 to 1000°C and pressure range of 1 to 10 kg per square centimeter absolute is discussed along with suitability of various gases as heat transfer media, behavior of fuel elements, and diffusion of fission products. (W.L.H.)

- 23 - 8 SUGGESTIONS FOR A HIGH TEMPERATURE PEBBLE PILE
Daniels, F.
(N-1668b) Chicago. Univ. Metallurgical Lab.
October 25, 1944. Decl. January 11, 1956. Contract (W-7401-eng.-37). (MUC-FD-8)

The possibility of building a pile of pebbles of uranium carbide and graphite, or equivalent materials, which will operate at 1500 and 2000°C is discussed. The pile is cooled either by circulating helium or boiling bismuth. The cooling gas passes uniformly through the whole cross-section of the pile, which is kept at a steady temperature by convection and radiation. Operation at a high temperature simplifies the cooling operation, opens up the possibility of separating Pu and fission products by vaporization, and leads to greater thermodynamic efficiency in the operation of engines. Experience gained during the recent operation of a gas-heated pebble-bed furnace blowing 2000 cu. ft. of air per minute for nitrogen-fixation, warrants the expectation that there are no serious practical difficulties in the operation of such a pile at 2000°C. (auth)

24. POWER PLANT

24 - 1 BUILDING A NUCLEAR POWER STATION

(Special Suppl. to British Engineering International).
May 1958. 18p.

The plant and equipment which have been installed at the Berkeley Nuclear Power Station are described. The Berkeley Station is taken as an example of the latest developments in the graphite-moderated, gas-cooled reactors for electricity generation; and the equipment described and illustrated is of the kind which could be supplied to a station of this type anywhere in the world. (W.D.M.)

24 - 2 CALDER HALL

Nuclear Power 1, 276-94(1956) November.

The power plant of Calder Hall is described including the description and uses of the turbines, steam cycle, steam control after a shutdown, normal steam dumping, controlled by oil operated valves connected to the pressure regulators and governors, faulty operation dumping by discharge to the atmosphere, water treatment to prevent scaling by ion exchange, demineralization and deaeration, and the hyperbolic reinforced concrete cooling towers. (F.S.)

24 - 3 CALDER HALL--FIRST LARGE-SCALE NUCLEAR POWER STATION

Eng & Boiler House, Rev. 71, No. 11, p. 354-65, November 1956.
see also Power & Works Eng., Vol. 51, No. 605, p. 401-4, November 1956.
Steam Engr., Vol. 26, No. 301, p. 47-53, November 1956.

Calder Hall "A" power station at Sellafield, Cumberland, opened Oct 1956, and identical station "B", to be completed in 1958, will ultimately comprise four reactors; two turbine houses each with four 23-Mw turboalternators; building and civil engineering work; reactors; heat exchangers; mounting and fittings; pipework; circulation; coolant gas system, etc.

24 - 4 CALDER HALL--THE FIRST NUCLEAR POWER STATION IN THE WORLD--HOW IT WAS BUILT--HOW IT WORKS

Nuclear Power 1, 240-58(1956) October.

The Calder Hall power station, consisting of two natural U, Pu producing graphite moderated, gas-cooled reactors in steel pressure vessels four turbo alternator turbines, two dump condensers for absorption of reactor heat output when turbines are out or on part load, and two hyperbolic cooling towers, is described. (F.S.)

24 - 5

EVALUATION OF FINNED TUBES FOR STEAM GENERATORS IN GAS COOLED POWER REACTOR SYSTEMS

Carr, F. (Nuclear Power Co., Ltd., Knutsford, Eng.),
Rounthwaite, C. and Young, P. A. (Head Wrightson and Co., Ltd.,
Thornaby, Eng.)
A/CONF.15/P/2327. 13p.

The power losses associated with the steam generators in a gas-cooled power reactor system are the gas side pumping and the finite temperature difference between the gas and water sides. A derivation is presented for reducing the power loss due to irreversible heat transfer to the basis applied to the pump power losses. An atmospheric air test rig of the closed circuit type and the associated experimental techniques are described. Experimental data are presented on heat transfer and pressure drop obtained from the extended surface tube used for the Bradwell power station. The correlation of the experimental results by some previously published methods is briefly discussed. (W.D.M.)

24 - 6

FIRST NUCLEAR POWER STATION

Commonwealth Engr., Vol. 42, No. 2, p. 57-9, September 1954.

Reactors and steam raising towers will replace conventional furnaces in Britain's first nuclear power station at Calder Hall, Cumberland; to obtain maximum efficiency from comparatively low temperature of gas entering steam towers, double pressure steam cycle is used; future large scale development of nuclear power.

24 - 7

HEAT RECOVERY FROM REACTORS

Chambadal, P., Rous, J. P.
Nuclear Power, Vol. 1, No. 5, p. 211-5, September 1956.
P. 266-9, October.

Four methods of steam raising are possible; promising method is that of using nuclear heat only for feedwater heating, with conventionally fired boiler supplying evaporation and superheat; it is shown that this is thermodynamically equivalent to continuous feed heating by bled steam; another method is by using flash steam boilers in cascade; description of power plants of French G1, G2 and G3 reactors of Marcoule.

24 - 8

INSTALLATIONS FOR ENERGY RECOVERY

Passerieux, P. and Scalliet, R.

Bull. inform. sci. et tech. No. 20, 99-114(1958) August. (In French)

The general characteristics of the water-steam circuits and of the electrical circuits are tabulated, and the layout of the installations for energy recovery is sketched. The installation for energy recovery has as its first function the proper cooling of the reactor under all conditions, and only secondarily the production of electricity. The means taken to ensure this operation are outlined. The heat exchangers, turbines, condensers, over-heating regulators, the control apparatus, and the pumps are described in detail. (J.S.R.)

24 - 9

RECOVERY OF THE ENERGY PRODUCED IN AIR COOLED GRAPHITE REACTOR G1.

Chambadal, P. and Pascal, M.

In International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Proceedings. Vol. 3, Power Reactors. New York, United Nations, 1956. p. 81-3

Design of an energy recovery system is described. Air coming out of the reactor at 220° goes thru an air-water exchanger raising the water to 200°. The water goes thru flash boilers which will supply steam to three stages of a turbine. The turbine will drive an alternating current generator developing 5750 kw, with 5500 v between terminals. The system was added after the reactor was designed and is intended primarily as a prototype.

24 - 10

STEAM BOILER FOR GAS COOLED PILE

Emerson, G. B.

(MonN-308) Clinton Labs., Oak Ridge, Tenn.

May 27, 1947. Decl. January 27, 1956. 21p. Contract W-35-058-eng-71.

The factors considered in arriving at selection of the type of steam boiler best suited to the requirements of the high-temperature power pile to be constructed at Clinton Laboratories are outlined. The sixteen principal factors considered in this connection are discussed in some detail. It is concluded that the forced recirculation type boiler is the most suitable type for this project. Preliminary specifications developed for this type of boiler are given in an appendix. (auth)

24 - 11

STEAM CYCLES FOR GAS-COOLED REACTORS

Wootton, W. R., Taylor, A. J. and Worley, N. G. (Babcock and Wilcox, Ltd., London).
A/CONF.15/P/273. 21p.

The potentialities of the steam cycle in association with possible trends in nuclear reactor development are investigated. Attention is confined to gas cooled reactors and emphasis is placed on the most probable coolant temperatures entering the reactor and those likely to be attained in the future at the reactor outlet. The coolant temperature leaving the reactor is assumed to progress in increment of 50°C in the course of time from 350 to 600°C and the most probable steam cycles are found for each stage of development. These lead to an indication of the trend of efficiency of future nuclear steam power plants of the gas-cooled type. (W.D.M.)

24 - 12

STEAM GENERATING SYSTEM AT CALDER HALL ATOMIC POWER STATION

Engineer, Vol. 197, No. 5132, 5134, p. 832, June 4, 1954, p. 902, June 18; see also Engineering, Vol. 177, no. 4612, p. 795, June 18, 1954. Eng & Boiler House Rev, Vol. 69, No. 7, p. 205-6, July 1954.

Reference made to nuclear power station in Cumberland, England; power station, under construction, will consist essentially of central turbine house with nuclear reactor structure situated separately at each end; heat from each reactor will be used to generate steam in four steam raising towers each 80 ft high; use of Babcock & Wilcox equipment.

25. PRESSURE VESSEL

25 - 1 BRADWELL REACTOR VESSELS. DESIGN, FABRICATION AND ERECTION PROBLEMS

Bishop, R. F. (Whessoe, Ltd.).
Nuclear Power 2, 406-9(1957) October.

Problems involved in the selection of a steel, which must be suitable for site-welding in thick plate form and for withstanding the high temperature working conditions, are summarized. Forming the plates, site prefabrication, erection inside the vault, and stress relieving and pressure testing are described also. (M.H.R.)

25 - 2 CALDER HALL---WELDED FABRICATION IN CONSTRUCTION OF BRITAIN'S FIRST ATOMIC POWER STATION.

Welding & Metal Fabrication, Vol. 24, No. 11, p. 384-92,
November 1956.

Design and construction of reactor pressure vessel; fabrication of five principal component parts; welding, oxygen cutting, stress relieving and test operations on pressure vessel; fabrication of heat exchanger towers; manufacture and erection of pipework.

25 - 3 CIVIL ENGINEERING OF THE PRESSURE VESSELS

Bellier, J. and Tourasse, M.
Bull. inform. sci. et tech. No. 20, 56-68 (1958) August (In French).

The pressure vessels of the reactors are the concrete shielding necessary for biological protection. To enable the concrete to withstand high pressures and stresses, concrete reinforced with steel cables is used. The advantages of the reinforcing and the general arrangement of the cables are discussed. A description of the pressure vessel construction and fabrication of the cables is given. The calculations made on the pressure vessels and the cables are reported. (J.S.R.)

25 - 4

IMPROVEMENTS IN OR RELATING TO PRESSURE VESSELS

Millar, R. N. and Mitchell, K. J. (to the General Electric Co., Ltd.).
British Patent 785,945. May 31, 1955.
Nuclear Power 3, 92(1958) February.

One design difficulty of the British type of gas-cooled reactor is in the design of the pressure vessel to withstand the pressure and heat for many years of continuous operation. Taken out by GEC's nuclear chief, R. N. Millar, and a colleague this patent shows how it is being tackled at the firm's Hunterston station. Briefly, the idea is to use two shells—one to withstand heat but little pressure, the other for pressure only. Five different forms of construction are described in the specification. Some fresh coolant gas is bypassed and distributed round the top part of the outer sphere and allowed to leak into the inner shell. By this type of construction, the outer shell can be ordinary boiler plate while the inner shell can be in relatively thin heat-resisting steel.

25 - 5

REACTOR PRESSURE VESSEL AND CHARGE MACHINE

Young, H. D. Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif.

In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 78-97.

A comparison is given of the dimensions of the pressure vessels for the nuclear power reactors in service and under construction in the United Kingdom and the vessel designs of KE-ACF. The prominent problems considered in the designs of the KE-ACF pressure vessels are discussed, including the selection of sizes and configurations, nuclear heating, induced radiation, radiation damage, brittle fracture probability, seismic stability, and requirements in addition to those of the ASME Code. The charge machine is described in some detail and its versatility is illustrated by a typical sequence of operations. (auth)

26. RADIATION DAMAGE

26 - 1

IRRADIATION DAMAGE AND ITS SIGNIFICANCE IN REACTOR DESIGN

Barnes, R. S. Sect. 2 (of) IRRADIATION DAMAGE.

(AERE-M and C/R-2751) United Kingdom Atomic Energy Authority.
Research Group. Atomic Energy Research Establishment, Harwell,
Berks, England.

April 2, 1958. 15p. Papers Presented at the First Colloquium of
the U.K.A.E.A. Diffraction Analysis Conference held at A.W.R.E.

Radiation damage in solids consisting of electronic displacements, atomic displacements, and impurity effects is briefly described. It is shown how the damage manifests itself in various components of the Calder Hall type of reactor. Components discussed include moderator, structural materials, fuel elements, and control rods. (W.D.M.)

27. REACTOR PHYSICS

- 27 - 1 ANALYSE SPECTRALE DES NEUTRONS RAPIDES A L'INTERIEUR DE BR-1
(Spectral Analysis of the Fast Neutrons Inside the BR-1 Reactor.)
Beets, C., Breny, H. and Fournaux, J. (Centre d'Etude de
l'Energie Nucléaire, Brussels).
A/CONF.15/P/1680, 38p.

The energy distribution of fast neutrons in a reactor is a fundamental datum which may be analyzed by nuclear emulsion beyond 250 kev. A measurement of two distributions at the interior of the BR-1 reactor was made at points of an elementary cell placed near the core of the reactor. One was at the center of the lattice in the graphite and the other was at a point near a rod. A detailed analysis made it possible to determine the corresponding neutron spectra from these results. (J.S.R.)

- 27 - 2 AN ANALYSIS OF POWER, TEMPERATURE AND REACTIVITY TRANSIENTS IN
B.E.P.O.
Wordsworth, D. V.
(AERE-E/R-1245) Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.
September 1953. Decl. March 1957. 21p.

A method is developed of calculating pile power and component temperatures in a pile of the BEPO type, following a sudden increase in reactivity by rapid withdrawal of the control rods. Comparison of three sets of experimental results, taken on BEPO with the corresponding theoretical results found by this method, shows that agreement is fair for the whole transient and is good for the points of maximum pile power and maximum U temperature, reached during the transient overshoots. It is suggested that time lags in recording pile power and temperatures account for some of the discrepancy between theory and experiment, and this is illustrated on the early part of the pile divergence where theory is well established. An over-all temperature coefficient of reactivity is found, suggesting that reactivity decreases 1.60 millimiles per degree rise in U maximum temperature. The results of the analysis confirm Sterry's heat transfer coefficients found on rig tests with BEPO helically finned slugs.
(auth)

27 - 3

BEPO WIGNER ENERGY RELEASE

Dickson, J. L., Kinchin, G. H., Jackson, R. F., Lomer, W. M. and Simmonds, J. H. W. (Atomic Energy Research Establishment, Harwell, Berks, Eng.).

A/CONF.15/P/1805. 97p.

The second Wigner energy release in the 6-Mw air-cooled graphite-moderated research reactor BEPO took place on 14-15th March 1958. The release was fully instrumented, and a preliminary analysis of the results shows the release to have been very complete. The reasons for the chosen method of release, the analysis and interpretation of the graphite sampling data from which the maximum anticipated release temperatures were predicted, and the various safety considerations of graphite oxidation and uranium cartridge failure under these maximum temperature conditions are described. The engineering arrangement for the conduct of the release are described followed by a description of the release and the temperature data obtained. (W.D.M.)

27 - 4

CHANNEL RATINGS FOR THE CALDER HALL REACTORS INCLUDING THE EFFECT OF XENON POISONING ON THE RADIAL FLUX DISTRIBUTION

Lang, J.M.B.

(RDB (R)/TN-56) United Kingdom Atomic Energy Authority. Industrial Group, H. Q., Risley, Lancs, England.

September 1, 1955. 11p. (TRSWP/R-16).

The channel ratings to be expected in the Calder Hall Thermal Reactors from flux values based on normal two-group calculations are compared with those modified to allow for Xe poisoning. (auth)

27 - 5

CLASSIFIED PROGRESS REPORT OF THE REACTOR SCIENCE AND ENGINEERING DEPARTMENT (FOR) APRIL 1-JUNE 30, 1950.

- - -

(BNL-67) Brookhaven National Lab., Upton, N. Y.

Decl. Feb. 14, 1957. 37p.

Neutron economy in the Brookhaven Reactor is considered as a function of core dimensions, reactor materials, loading, and other criticality factors.

27 - 6 CLASSIFIED PROGRESS REPORT OF THE REACTOR SCIENCE AND ENGINEERING
DEPARTMENT JANUARY 1-MARCH 31, 1950

(BNL-52) Brookhaven National Lab., Upton, N. Y.
Decl. February 14, 1957. 43 p.

Subcritical Experiments on BNL Reactor. During the report period it was possible to load the reactor to give an effective multiplication constant of 0.99. The incomplete evaluation of the data found to date indicates a flux distribution with a reflector saving of 62 cm, in agreement with two-group theory. The effect of the central gap for admission of cooling air appears to lie within the limits for the calculated values. The critical loading for the reactor is given.

27 - 7 A DETERMINATION OF THE LAPLACIANS IN GLEEP

Littler, D. J. and Stewart, J. C.
(AERE-N/R-131) Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.
September 16, 1947. Decl. July 24, 1958. 15p.

An attempt has been made to determine the laplacians in Gleep by exploring the thermal-neutron flux distribution with indium foils. The laplacian of the uranium metal-graphite lattice is $0.92 \pm .04\text{m}^{-2}$ and the laplacian of the uranium dioxide-graphite lattice is $0.66 \pm 0.4\text{m}^{-2}$.
(auth)

27 - 8 A DETERMINATION OF THE RATE OF EMISSION OF SPONTANEOUS FISSION
NEUTRONS BY NATURAL URANIUM

Littler, D. J.
Proc. Phys. Soc. (London) A. 65:203-8, (March 1952).

By comparison with a calibrated neutron source, the total rate of emission of primary neutrons in the Graphite Low Energy Experimental Pile has been found. The rate of emission of spontaneous fission neutrons from natural U is computed to be 59.5 ± 3.3 neutrons/g/hr of U.

27 - 9 DISTRIBUTION SPECTRALE DES NEUTRONS LENTS DE BR-1. (Spectral
Distribution of Slow Neutrons in BR-1.)

Neve de Meyergnies
(BLG-26) Brussels. Centre d'Étude de l'Énergie Nucléaire.
April 17, 1958. 32p.

In a weakly absorbing medium, the slow neutron spectrum can be represented with a good approximation by the sum of a Maxwellian distribution and of a resonance distribution following the $E^{-3/2}$ law. Because of the hardening of the spectrum by absorption, the temperature T_M of the almost Maxwellian part does not correspond to the physical temperature T_p of the medium. The experimental results indicate the following equation for graphite: $T_M/T_p = 1 + (3.36 \pm 0.33) \times \phi_{epi}/\phi_{th} \times T_o/T_p$. (auth)

27 - 10 DIVERGENCY DATA FOR GLEEP

Watson-Munroe, C. N.

(AERE-N/R-121) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.

August 22, 1947. Decl. July 24, 1958. 9p.

The methods employed for measuring neutron fluxes and estimating the critical radius for the Gleep (Graphite Low Energy Experimental Pile) are described. Descriptions of the methods of loading and details of the experimental data are given. (auth)

27 - 11 EFFECT OF PRESSURE, TEMPERATURE AND POISON ON THE REACTIVITY OF THE HARWELL PILE

Fenning, F. W.

(POC/Mem-20) (Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.)

February 1, 1949. 3p.

The loss in k during operation of a pile due to rise in temperature and accumulation of neutron absorbing fission products is known to be quite large and an attempt has been made to analyse the measured losses on the Harwell Pile in terms of temperature coefficients and "poisoning" by Xe^{135} . The variation of the latter with time has been computed by the Theoretical Division on the assumption that all the Xe^{135} is produced from the Te^{135} chain of fission products. (auth)

27 - 12 EFFECTS ON REACTIVITY OF EXPERIMENTAL ABSORBERS

Auerbach, T.

(BNL-3348) Brookhaven National Lab., Upton, N. Y.

November 9, 1956. 17p.

The calculations used to predict the reactivity effects produced by inserting an absorber into an experimental hole of the BNL reactor are presented and discussed. The calculations are considered applicable to weak absorbers only. The specific case of a U-Bi radiation effects loop is treated. (D.E.B.)

- 27 - 13 ETUDE D'UN SYSTEME URANIUM ENRICHI-GRAPHITE REFROIDI PAR AIR
(Study of an Enriched Uranium--Graphite System Cooled by Air).
Dopchie, H.
(NP-6278) Brussels. Centre D'Etudes Pour Les Applications de
l'Energie Nucleaire.
April 13, 1956. 9p. (B-3).

A study was made of the introduction of elements containing 20% enriched U in the BR-1 reactor. These elements are formed by three curved Al plates containing the enriched U and joined to each other by Al rings. They are similar to the elements described in report P/860 of the Geneva Conference entitled "The Brookhaven Reactor" by Marvin Fox. It is shown that if one limits the maximum air speed to 100 m/sec, the exit air temperature to 300°C, and the plate temperature to 350°C, the maximum allowable flux is about 10^{13} n/cm²sec. It is also shown that this allowable flux is practically independent of the amount of U enrichment, above a value of 20% in U²³⁵. (tr-auth)

- 27 - 14 AN EXPERIMENTAL AND THEORETICAL STUDY OF THE SUBCRITICAL BNL REACTOR
Chernick, J., Kaplan, I., Kunstadter, J., Sailor, V. and Williams, C.
(BNL-60) Brookhaven National Lab., Upton, N. Y.
June 15, 1950. Decl. December 14, 1955. 109p.

Experimental and theoretical studies of the subcritical BNL Reactor are summarized. Subcritical and exponential experiments are described. Experiments were also performed on transients and the effects of displacement of U. Theoretical consideration was given to lattice calculations, anisotropic effects, neutron leakage from the gap, reflector saving, and other reactor constants.
(B.J.H.)

- 27 - 15 EXPONENTIAL EXPERIMENTS FOR P.I.P.P.A. MK. 1.
Bevan, G., Mummery, P. W., Salmon, A. J. and Thompson, M. W.
(AERE-R/R-2602) United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.
1958. MS. Date November 1954. Decl. June 1958. 76p.

Declassified version of AERE-RP/R-1545.

A series of exponential experiments were performed which were designed to provide criticality data for PIPPA Mark I and to systematically increase the amount of criticality

data for gas-cooled graphite-moderated uranium reactors in general. The experimental methods used and the method of analysis of the observations are described in detail. Tables of experimental observations are given together with the values of the Laplacian and the migration area asymmetry deduced from the analysis. (auth)

27 - 16

EXPONENTIAL EXPERIMENTS WITH NATURAL AND SLIGHTLY ENRICHED URANIUM, GRAPHITE MODERATED SYSTEMS

Lockey, D. J., Pearson, H., Pope, A. L. and Wells, G. M. (AERE-R/R-2597) United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.
September 1956. Decl. May 1958. 54p.

Declassified version of AERE-R/R-2014.

An account is given of graphite exponential experiments which extend the range covered previously. An improved method of analysis is employed. The results are compared with the lattice calculations recommended for PIPPA type reactors, and proposals are made for further work. (auth)

27 - 17

FAST-NEUTRON FLUX MEASUREMENTS IN THE ORNL GRAPHITE REACTOR SHIELD

Blosser, T. V., Bond, G. W., Lee, L. A., Morgan, D. T., Nichols, J. F., Reid, R. C., Reynolds, A. B., Speidel, T. O. P., Vroom, D. W. and Welt, M. A.
(CF-58-8-1) Oak Ridge National Lab., Tenn.
August 14, 1958. 11p. Contract (W-7405-eng-26).

In a study of the nuclear and physical properties of the concrete shield of the ORNL Graphite Reactor measurements were made of the fast-neutron fluxes in a hole through the shield. These fluxes were not included in the formal report of the study (ORNL-2195) since several uncertainties associated with the measurements made the absolute values unreliable. The attenuation curves at one energy may be reasonably accurate, however. (auth)

27 - 18

FLUX CHANGES WITH LONG-TERM IRRADIATION IN A REACTOR OF THE CALDER HALL TYPE

Hitchcock, A. (U.K.A.E.A. (I.G.), Risley, Warrington); Price, V. E. (English Electric Co., London); and Shenton, Y. (Metropolitan Vickers Electrical Co., Ltd., Trafford Park, Manchester).
J. Nuclear Energy 6, 66-78(1957).

A method has been developed for evaluating flux and reactivity changes in a reactor due to long-term changes in the

quality of the fuel under irradiation. An account is given of the method, and the results in two simple cases. It is tentatively concluded that the permissible maximum heat output and channel ratings may vary very considerably during the lifetime of a charge, a point of possible importance in reactor design. (auth)

- 27 - 19 FLUX FINE STRUCTURE IN GAS COOLED REACTOR CELLS
Riesch, G. (AEG-Kernenergieanlagen, Frankfurt am Main).
Atomkern-Energie 3, 16-21(1958) January. (In German)

The calculations of Newmarch on the flux fine structure in a cylindrical cell with gas cooled central fuel rods were expanded for bundles of fuel rods. (J.S.R.)

- 27 - 20 HARWELL WIGNER RELEASE, FIRST SINCE WIND-SCALE

Nuclear Power 3, 230(1958) May.

A new method of releasing stored energy was recently carried out on BEPO at Harwell. Instead of making the reactor critical with no coolant flow, the air was preheated, while the reactor was kept running at a very low power level. The release was started by switching on 750kw of heating grids installed in the air inlet duct and reducing the flow to 5% of normal. The graphite was heated to 90°C where the temperature then jumped up in some places by as much as 200°C in five minutes. The maximum temperature reached in the graphite was 330°C. After 26 hr the temperature was falling steadily in all parts of the reactor, and the release was considered at an end. (W.D.M.)

- 27 - 21 THE MEASUREMENT OF AN ABSOLUTE THERMAL NEUTRON DENSITY IN G.L.E.E.P.
Littler, D. J. and Lockett, E. E.
(AERE-R/R-961) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.
September 5, 1952. 15p.

By measuring the thermal neutron absorption cross-section of Na on the G.L.E.E.P. oscillator, and by irradiating samples of Na₂ CO₃ at the center of G.L.E.E.P., and subsequently determining the absolute B-activity of these samples, an absolute thermal neutron density at the center of the pile has been measured. This neutron density can

be reproduced accurately by measuring the current from a boron coated ion chamber situated near the center of the pile. The density is at present expressed in terms of the thermal neutron absorption cross-section of a particular sample of boron. An absolute measurement of this cross-section is being made, but has not yet been completed. The accuracy of the density determination is $\pm 4\%$, but this error will have to be increased statistically by the error on the determination of the boron cross-section. The thermal neutron density which can be provided covers the range of approximately 10 to 2×10^4 n/cm³.

27 - 22

MEASUREMENTS ON A SUBCRITICAL REACTOR USING A PULSED NEUTRON SOURCE
Sjostrand, N. G.
Arkiv Fysik II, 233-46(1956)

A pulsed neutron source was introduced into the Swedish heavy-water reactor R1 when it was in a subcritical state. The time behavior of the neutron density was studied by means of a multichannel time analyzer. Using this technique the time constant of the prompt neutrons and the reactivity was determined at several heavy water levels down to about 20 cm below the critical level. It was found that the reactivity equivalent of the heavy water is $(7.2 \pm 0.1) \times 10^{-6}$ per mm, and that the mean lifetime of the thermal neutrons in the reactor is 0.68 ± 0.02 msec. These values refer to the critical level and are based on a fraction of delayed and photo neutrons amounting to 0.78×10^{-2} . A calibration of the safety rods gave the result that each of them corresponds to a reactivity of 1.7 ± 0.2 per cent and both together to 3.4 ± 0.4 per cent. (auth)

27 - 23

MEASURING REACTOR SPECTRA WITH THRESHOLDS AND RESONANCES
Trice, J. B. (Oak Ridge National Lab., Tenn.).
Nucleonics 16, No. 7, 81-3(1958) July.

Fission probes and activation detectors were used to measure and compare the neutron spectra in the Bulk Shielding Reactor, the Low Intensity Test Reactor, the ORNL Graphite Reactor, and the Materials Testing Reactor. The method used (utilization of the resonance and threshold reactions) is sensitive to cross-section data and can become an incisive approach to knowledge of reactor spectra. (J.S.R.)

- 27 - 24 METHOD OF SUSTAINING A NEUTRONIC CHAIN REACTING SYSTEM
Fermi, E. and Leverett, M. C. (to U. S. Atomic Energy
Commission).
U. S. Patent 2,813,070. November 12, 1957.

This patent relates to neutronic reactors and a method of sustaining a chain reaction. The reactor shown in the patent for carrying out the method is the gas-cooled type comprised of a solid moderator having a plurality of passages therethrough for receiving bodies of fissionable material. In carrying out the method, the reactor is loaded by inserting in the passages fuel elements and moderator material in a proportion to sustain a chain reaction. As the reproduction ratio decreases below the desired figure due to impurities formed during operation of the reactor, the moderator material is gradually replaced with additional fuel material to maintain the reproduction ratio above unity.

- 27 - 25 A MONTE CARLO ESTIMATION OF THE HIGH ENERGY NEUTRON FLUX
DISTRIBUTION IN THE ORNL GRAPHITE REACTOR.
Moshman, J.
(ORNL-1836) Oak Ridge National Lab., Tenn.
February 23, 1955. Decl. March 1, 1957. 32p. Contract
W-7405-eng-26.

Three representative regions within the ORNL graphite reactor lattice were chosen for investigation of the fast neutron flux. GPC and UNIVAC calculations were carried out for 14 energy intervals between 0.025 and 10 Mev. (K.S.)

- 27 - 26 NEUTRON STREAMING IN GAS-COOLED REACTORS
Grant, I. S.
(AERE-R/R-2568) United Kingdom Atomic Energy Authority.
Research Group. Atomic Energy Research Establishment, Harwell,
Berks, England.
October 1958. 25p.

Calculation of the leakage in reactor lattices containing cooling channels is discussed in terms of diffusion theory. Anisotropic diffusion coefficients are proposed which make allowance for the effect of the fine structure of the neutron distribution across a lattice cell. (auth)

- 27 - 27 NEUTRON STUDY OF THE G-2 AND G-3
Teste du Bailler, A.
Bull. inform. sci. et tech. No. 20, 1-9(1958) August. (In French)

The essential characteristics, as well as the principal numerical values, of the natural uranium-graphite matrix of the G-2 reactor are given. The calculation methods and the formulas used are not reproduced. The reactivity control and the flattening of the flux curve are briefly considered. (J.S.R.)

- 27 - 28 PHYSICS OF THE PROTOTYPE ENRICHED GCPR
Wood, P. M. (Nuclear Products-Erco Div.), ACF Industries, Inc.
Washington, D. C.
In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 49-53.

In the physics analysis of the prototype Enriched Gas Cooled Power Reactor the usual calculations of lattice parameters, reactivity lifetime, and control rod worth were made. Some of the results which most affect the operating characteristics of the core are presented. (auth)

- 27 - 29 THE PRODUCTION OF THRESHOLD REACTIONS IN A GRAPHITE REACTOR
Mellish, C. E., Payne, J. A. and Otlet, R. L.
In International Conference on Radioisotopes in Scientific Research, Paris, Proceedings. Pergamon Press, Ltd., 1957. Vol. 1, p. 35-49.

The reaction $Ni^{58}(n,p)Co^{58}$ has been used to monitor the flux of fast fission neutrons in the Harwell reactor BEPO in a series of different positions in the reactor, making assumptions about the energy spectrum of the flux. The ratio of fast to slow flux is plotted through the reactor, and this gives the preferred positions for producing threshold reactions and the preferred positions for eliminating them as far as possible. The importance of this elimination in some radioactivation analyses is pointed out. The occurrence of "exothermic" (n,p) reactions such as $N^{14}(n,p)C^{14}$ and $Cl^{35}(n,p)S^{35}$ is discussed. Cross sections are presented for a number of (n,p) reactions on elements ranging from Ti to Tl in the Periodic Table, and for some (n,α) reactions on elements from Ti to Cs. Where the product isotope can also be formed by neutron capture in impurity a method is devised to correct for any impurity present. The cross sections measured are compared with those predicted by theory. (auth)

27 - 30

REACTOR PHYSICS MEASUREMENTS DURING THE COMMISSIONING OF THE CALDER REACTORS

Leslie, F. M. (United Kingdom Atomic Energy Authority, Sellafield, Cumb., Eng.)

A/CONF.15/P/315. 11p.

Details of some of the measurements made during the commissioning and early operation of the Calder reactors are presented. The measurements include critical size, lattice constants, control rod calibration, xenon poisoning, temperature coefficients, and reactivity build-up. Brief mention is made of future experimental work on control rods and temperature coefficients. (auth)

27 - 31

REACTOR SCIENCE AND ENGINEERING DEPARTMENT PROGRESS REPORT FOR JANUARY 1—JUNE 30, 1951.

(BNL-113) Brookhaven National Lab., Upton, N. Y.

Decl. February 14, 1957. p. 29-47.

Various measurements on BNL reactor constants made during a start-up are discussed, most of them agreeing well with theoretical design predictions. An experiment allowing the reactor to reach low power at no cooling is described. The temperature of the metal went through several thermal oscillations before coming to equilibrium, and equations are derived to describe the observed effect. (D.E.B.)

27 - 32

SLOW-NEUTRON SPECTRUM MEASUREMENTS FROM THE SWEDISH HEAVY-WATER REACTOR, FL.

Larsson, K. E., Stedman, R. and Palevsky, H. (AB Atomenergi, Stockholm).

J. Nuclear Energy 6, 222-32(1958).

During the last few months a slow chopper with associated time-of-flight equipment has been put into operation at the Swedish reactor, FL. The first task was to determine the shape of the slow neutron spectrum in the wavelength region from 0.85 to 11 Å. The spectrum of the slow neutrons emerging from an experimental hole is best fitted with a function which is nearly Maxwellian in shape $dn = \text{const. } \lambda^{4.07} \exp[-(\lambda_0/\lambda)^2] d\lambda$, where $\lambda_0 = 1.66_8 \text{ Å}$ and corresponds to a neutron temperature of $(341 \pm 10)^\circ \text{K}$. The moderator temperature on the same occasion was 298°K . In subsidiary measurements it has been shown that the percentage of neutrons below the Be cut-off at 4 Å is 5% of the total slow neutron spectrum, in agreement with the velocity distribution data. At a reactor power of 600 kw

the flux of thermal neutrons below the cadmium cut-off at the exit of this experimental channel of radius 12 cm is 3.4×10^8 n/cm² sec, the flux below the Be cut-off being about 1.3% of this or 4.4×10^6 n/cm² sec. (auth)

27 - 33

SOME METHODS OF CALCULATING NUCLEAR PARAMETERS FOR HETEROGENEOUS SYSTEMS

Barden, S. E., Blackburn, D., Keller, A. Z. and Watson, J. M. R. (AEI-John Thompson Nuclear Energy Co., Ltd., Knutsford, Ches., Eng.). A/CONF.15/P/272. 30p.

Developments and applications of the Wigner-Seitz method and the individual sink and source method for calculating the relevant parameters of heterogeneous systems are discussed with particular reference to uranium-fueled, gas-cooled, graphite-moderated reactors. Methods suitable for general geometries of fuel and channel are developed. Numerical results are obtained for some rectangular channels using the Wigner-Seitz method and for the amount of reactivity taken up by an array of control rods by using the sink and source method. A common starting point for the generalization of both methods can be based on a single integro-differential equation which is formulated and discussed. (W.D.M.)

27 - 34

SPECTRUM OF NEUTRONS FROM THE SACLAY PILE

Galula, M., Jacrot, B. and Netter, F.
Letter in J. Phys. Radium, Vol. 16, No. 3, 232-3 (March 1955)
(In French.)

Results of time-of-flight spectrometer measurements are reported.

27 - 35

STUDY OF THE MULTIPLICATION FACTOR IN THE SACLAY REACTOR

Jacrot, B., Netter, F. and Radievski, V.
J. Phys. Radium, Vol. 15, Suppl. No. 1, 31-38A (January 1954)
(In French).

Methods of determining the effective multiplication factor in a reactor are discussed. The theoretical relationship between this factor and the heavy water level in a cylindrical pile is set out and the observed behavior of the Sacalay reactor is shown to be in accordance with the theory. The methods are applied in determining the effectiveness of control of safety rods, the effect on reactivity of the nitrogen coolant and in the evaluation of the power level of the reactor.

27 - 36

STUDY OF THE NEUTRONICS OF TWO FRENCH HEAVY WATER REACTORS

Horowitz, J.

In International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Proceedings. Vol. 5: Physics of Reactor Design. New York, United Nations, 1956. p. 256-267.

The two French reactors - the Chatillon reactor, called ZOE, and the Saclay reactor, called P-2 — have been the object of detailed neutronic studies. These studies include macroscopic distribution of the neutron flux: comparisons between the Laplacians of the lattices, variation of the reactivity with heavy-water level, experimental verification of the Nordheim formula, average life of the neutrons in a reactor, neutron spectrum of the Saclay reactor, effects of temperature on reactivity, and the xenon effect.

27 - 37

A SURVEY OF LATTICE CALCULATIONS FOR GRAPHITE MODERATED GAS COOLED THERMAL REACTORS

Cutts, B. ed.

(RDB(R)/TN-55) United Kingdom Atomic Energy Authority. Industrial Group H. Q., Risley, Lancs, England. September 1955. 50p. (NPCC/RPWP/P-6).

The edited results of a wide range of lattice calculations are given which can be used in solving several kinds of reactor problems. In particular the critical size of a natural uranium reactor may be found or alternatively the critical enrichment for any given reactor size. The important lattice constants for an infinite reactor without poisoning are summarized. (auth)

27 - 38

THE SWEDISH HEAVY WATER REACTOR—TWO KINETIC EXPERIMENTS

Hellstrand, E.

Atomica, Vol. 6, No. 7, 187-94, (July 1955).

The prompt neutron lifetime in the Swedish heavy water reactor was determined by two standard methods, the first involving analysis of phase and amplitude modulation of the power level resulting from a periodic modulation of reactivity, the second involving analysis of the time variation in power following a step change in reactivity. Good agreement was obtained, an effective mean lifetime of neutrons in the reactor medium of 0.71 ± 0.03 msec. being obtained.

- 27 - 39 THERMAL NEUTRON ABSORPTION CROSS-SECTION OF BORON
Green, A., Littler, D. J., Lockett, E. E., Small, V. G. Spurway,
A. H. and Bowell, E.
J Nuclear Energy, Vol. 1, No. 2, p. 144-9, December 1954.

Use of boron as standard for absorption cross section measurements made on GLEEP oscillator; measurements of thermal neutron absorption cross section of boron made at Harwell, England, in Sweden, and United States; agreement between Harwell and Swedish values is good, but there is disagreement with latest American values. Bibliography.

- 27 - 40 THE THERMAL-NEUTRON FLUX DISTRIBUTION IN GLEEP, TOGETHER WITH A
DEFINITION OF THE GLEEP UNIT OF POWER
Lockett, E. and Stewart, J. C. C.
(AERE-N/R-162) Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.
January 30, 1948. Decl. April 18, 1958. 13p.

Detailed information on the thermal flux available throughout the experimental holes of GLEEP is presented. (D.E.B.)

- 27 - 41 THE TRANSMISSION OF THERMAL NEUTRONS IN A STRAIGHT ANNULAR DUCT
Horton, C. C., Halliday, D. B., Lakey, J. R. A. and Harrison,
J. R.
(SWP/P-27) Gt. Brit. Atomic Energy Research Establishment, Harwell,
Berks, England.
January 1956. 10p.

The transmission of thermal neutrons in straight annular ducts through the BEPO reactor shield has been investigated over the range $1 < x/R_{\text{eff}} < 60$, where R_{eff} is related to the cross-sectional area of the duct by the expression $\pi R_{\text{eff}}^2 = A$ and x is the distance along the duct. It is found the results can be fitted by a theoretical formula using a calculated value of the albedo for the duct wall. (auth)

- 27 - 42 TWO GROUP CORE CALCULATIONS FOR THE CALDER HALL THERMAL REACTORS
Cutts, B.
(RDE(R)/TN-43) Gt. Brit. Dept. of Atomic Energy. Industrial Group
H. Q., Risley, Lancs, England.
June 2, 1954. 16p. (TRDC/P-42).

Detailed calculations of the nuclear properties of the core of a Calder Hall Thermal Reactor have been made. The zones of different channel size have been allowed for, but the

Wigner gaps have been assumed to have a uniform size throughout the core. The different properties of graphite in the core and reflector sections around and at the ends of the core have been considered. The quantity and enrichment of U required to fill an outer zone in order that the pile shall just run at 180 Mw heat rating, when the inner zone is loaded with natural U, is given for inner zone radii from zero to 12 ft. (auth)

27 - 43

THE USE OF ENRICHED URANIUM IN B.E.P.O.

Fenning, F. W.

(POC/Mem-37) (Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.)

May 19, 1950. Decl. July 4, 1957. 6p.

The possibility of replacing part or all the U in B.E.P.O. with ^{235}U -enriched U in order to increase the available excess k has been suggested. The time scale for production of enriched metal is such that it is desirable to proceed with the second charge with natural U and replace a part of this when enriched material becomes available. A preliminary theoretical investigation was made to find the degree of enrichment required to gain a few % in k for (a) 20 control channels replaced by enriched material of 0.8 in. diameter, (b) 40 central channels replaced by enriched material of 0.8 in. diameter, (c) complete charge enriched, and 0.9 in. diameter metal retained. The neutron flux distributions for (a) and (b) were examined to determine the maximum/average heat output per cartridge. In order to reduce this the metal diameter considered for the central region was 0.8 in. (instead of 0.9 in.). A reduction in diameter is also justified from the effect of enrichment on the rod diameter for optimum k in a fixed lattice. (auth)

27 - 44

THE USE OF INDUCED N^{16} ACTIVITY IN A REACTOR COOLANT FOR THE DETERMINATION OF NEUTRON FLUX DISTRIBUTION

Goodings, A.

(AERE-EL/M-105) United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England.

1958. 14p.

The possibility of using induced coolant activity to study the integrated neutron flux along a fuel element channel is discussed. It is concluded that the technique may be useful, and possible lines of further study are proposed. (auth)

THE USE OF THE PILE OSCILLATOR IN THERMAL REACTOR PROBLEMS

Rose, H., Cooper, W. A. and Tattersall, R. B. (Atomic Energy Research Establishment, Harwell, Berks, Eng.).

A/CONF.15/P/14. 26p.

An experimental program was carried out at Harwell, using the pile oscillator facilities of Dimple and Gleep. Thermal and epithermal absorption effects were separated by integral measurements in several different neutron spectra. A large number of samples were studied, including typical casting and structural materials, the heavy nuclei Np^{237} and Pu^{240} , and many of the important fission product poisons. The fissile nuclei U^{235} , U^{233} , and Pu^{239} were compared, and relative η -values deduced for several spectrum conditions. The Gleep pile oscillator was also used to make measurements of the changes in cross section of nuclear fuels arising from intensive neutron irradiation. Data were obtained for a variety of NRX and Windscale irradiated samples including natural uranium, thorium, and aluminum alloys containing small amounts of pure fissile material. (auth)

28. SAFETY

28 - 1 ACCIDENT AT WINDSCALE

Power Eng., Vol. 62, No. 2, p 45-7, February 1958.

Report on nuclear accident at Britain's Windscale No. 1 Pile, October 10, 1957, which occurred during maintenance operation called Wigner Release; circumstances and damage during nuclear heating leading to accident; two separate and distinct faults of instrumentation affecting second nuclear heating; degree of contamination.

28 - 2 THE ACCIDENT AT WINDSCALE AND THE RADIATION DAMAGE TO GRAPHITE Gandusio, G. and Polezzo, S. (ENI, Milan). Energia nucleare (Milan) 5, 289-96(1958) May. (In Italian)

The causes and effects of the Windscale accident are discussed. The problems resulting from such an accident are reviewed with special attention to the Wigner release and the structural modifications of graphite caused by the irradiation. (J.S.R.)

28 - 3 ACTIVITEIT VERSPREID IN DE OMGEVING BIJ NORMAAL BEDRIJF VAN DE BR1. (Activity Scattered in the Surrounding in Normal Operation of the BR-1.) Vandewiele, L. (NP-7246) Brussels. Centre d'Etudes Pour Les Applications de l'Energie Nucleaire. December 23, 1957. 12p.

Calculations were made of the radioactivity in the air coolant and of the gamma doses and ground concentrations during normal operation of the BR-1. The results are given. (J.S.R.)

28 - 4 AN ANALYSIS OF THE MAGNESIUM-CO₂ REACTIONS IN TERMS OF A REACTOR ENVIRONMENT. Scott, J. L. (CF-58-5-100) Oak Ridge National Lab., Tenn. May 15, 1958. 19p. Contract (W-7405-eng-26).

The hazards associated with the use of magnesium capsules in a CO₂-cooled reactor system have been analyzed, and

consideration has been given to incidents which could lead to excessive temperature excursions and factors which could significantly reduce the ignition temperature of magnesium alloys. Methods of reducing the hazards associated with the Calder Hall type of reactor are discussed, and an experimental approach with which to define more accurately conditions that would lead to ignition is suggested. (auth)

28 - 5

THE ATMOSPHERIC DIFFUSION OF GASES DISCHARGED FROM THE CHIMNEY OF THE HARWELL REACTOR BEPO

Stewart, N. G., Gale, H. J. and Crooks, R. N. (Atomic Energy Research Establishment, Harwell, Berks, Eng.)

Intern. J. Air Pollution 1, 87-102(1958) October.

The 61 m chimney of the BEPO reactor discharges warm air containing a small amount of radioactive argon. This gas has been used as a tracer to study the diffusion of the plume in the atmosphere downwind of the chimney. Surveys have been carried out at ground level up to a distance of 10,000 m and, with instruments suspended from a balloon, up to a height of 300 m at a maximum range of 1000 m. The results have been correlated with the gradient of potential temperature in the atmosphere near ground level. The effective height of the plume is found to depend on the wind velocity u and on the temperature gradient, having a mean value of 125 m under adiabatic conditions when $u = 7$ m/sec. The diffusion of the plume over the range studies occurs in three distinct phases. Near the chimney, the rate of vertical diffusion is about twice that in the horizontal direction as a result of the discharge conditions. As the plume diffuses towards ground level it enters a region dominated by building turbulence and the rate of diffusion in both directions is greatly increased. Finally, as it travels beyond the build-up area, the rate of diffusion reverts to that characteristic of diffusion over open country. Sutton's diffusion equation cannot be simply applied to this complex type of problem, but they provide a useful model for classifying the data, and effective values of the virtual diffusion constants C_y and C_z are derived for each stage of the diffusion process. Under adiabatic and stable conditions, peak ground-level concentrations between 1.3×10^{-6} and 3.5×10^{-6} units/m³ were observed at approximately 1000 m downwind with wind velocities in the range 6 to 12 m/sec and a chimney emission rate of 1 unit/sec. Under statically stable conditions no detectable concentrations were found on the ground within 10,000 m of the chimney. (auth)

- 28 - 6 CALDER WORKS: THE RADIOLOGICAL DESIGN OF THE DISCHARGE PLANT
Dent, K. H. and Knights, H. C.
(IGE-R-7) United Kingdom Atomic Energy Authority. Industrial
Group H. Q., Risley, Lancs, England.
1958. 14p. (IGC-TROC/R-4; IGC-TRDC/P-139).

The general basis of the radiological design of the plant is described, and dose rates are indicated for each operation together with annual integrated doses for each key operator. (auth)

- 28 - 7 EXPERIMENTAL DETERMINATION OF THERMAL TIME LAGS OF THE HARWELL
PILE AND THEIR EFFECT ON PILE SAFETY
Moore, R. V.
(POC/Mem-32) Gt. Brit. Atomic Energy Research Establishment,
Harwell, Berks, England.
August 26, 1949. Decl. July 4, 1957. 19p.

Empirical equations are constructed from test data, expressing the rise in uranium, graphite and channel air temperature with time for a suddenly generated nuclear power and comparison is made with equations deduced theoretically. Similar expressions are developed for a pile diverging exponentially with a given period and the effectiveness of ion chamber and thermal trips of the Harwell Pile assessed for a 10 second doubling time. The limit to which the power of a diverging pile may be considered exponential is investigated for a 20 second period and this data also illustrates the stabilizing effect of the uranium temperature coefficient. The effect of thermal time lags on operational temperature and power measurement is included. (auth)

- 28 - 8 HAZARDS SUMMARY REPORT FOR THE GCRE CRITICAL-ASSEMBLY EXPERIMENTS
Hogan, W. S., Dingee, D. A., Jankowski, F. J. and Chastain,
J. W.
(BMI-1240) (Rev.) Battelle Memorial Inst., Columbus, Ohio.
July 22, 1958. 88p. Contract W-7405-eng-92.

Critical experiments are described, and the hazards attendant to these experiments are evaluated for a gas-cooled water-moderated reactor design. Shutdown control of the critical assembly is achieved by cadmium control blades and by dumping the water into a storage tank. The maximum normal rate of reactivity addition is limited to an estimated 0.04 per cent per sec for all remotely controlled operations. The system is interlocked to insure that safe operational procedures are followed. Hazard calculations

are made to determine the dosage from direct radiation, fall-out, and inhalation from a radioactive cloud resulting from an accident. The exclusion area is shown to be adequate for even the maximum hypothetical accident. (auth)

28 - 9

POSSIBLE MECHANISM OF FAILURE OF CARBON DIOXIDE COOLED REACTOR
PRESSURE VESSELS

Heal, T. J., Watkins, B. and Cowan, A.
(IGR-TN/C-632) .Gt. Brit. Culcheth Labs., Culcheth, Lancs, England.
October 1957. 7p.

The factors likely to cause failure of reactor pressure vessels are discussed, distinguishing those which may occur prior to service from those which may occur during service. It is concluded that no single factor is of paramount importance. Failure during service would cause greater concern and, accordingly, overriding consideration may have to be given to factors (creep, high temperature ductility, metallurgical stability, effects of irradiation and fatigue) likely to lead to failure at this stage. (auth)

28 - 10

REACTOR HAZARDS

Perry, F. I. (Nuclear Products-Erco Div.), ACF Industries, Inc., Washington, D. C.
In Information Meeting On Gas-Cooled Power Reactors, Oak Ridge National Laboratory, (TID-7564), October 21-22, 1958. p. 70-7.

The hazards resulting from various assumed accidents and component failures were examined for the prototype, natural uranium and partially enriched uranium gas cooled power reactor systems. In general, the hazards associated with both types of reactors are similar. Both systems are characterized by large core heat capacities and relatively slow control rod withdrawal rates. These characteristics combined with the prompt negative temperature coefficient of reactivity of the fuel makes both reactors stable to power and temperature transients. The major difference in the two reactor systems is the presence of the chemically reactive fuel element materials, metallic uranium and Magnox, in the natural uranium system. (auth)

28 - 11

REPORT ON THE BROOKHAVEN NUCLEAR REACTOR PREPARED FOR REACTOR SAFEGUARD
COMMITTEE OF THE ATOMIC ENERGY COMMISSION

Borst, L. B.
(BNL-18) Brookhaven National Lab., Upton, N. Y.
June 22, 1948. Decl. September 19, 1956. 186p.

General design features of the Brookhaven Reactor are described. Photographs of the site and partially completed buildings are included. Tabulated data compare the Brookhaven and the Clinton reactors. Detailed information is presented on the following features of the Brookhaven Reactor: nuclear characteristics; startup and shutdown; control rod drive equipment; reactor instrumentation; metal protection; equipment and operation of the helium system; efficiency and servicing of the filter system; metal storage in and contamination of the canal; the cooling system; the electrical power system; meteorology and radiation protection; and a survey of natural catastrophies as a possible source of trouble. (C.H.)

- 28 - 12 SAFETY ASPECTS OF HETEROGENEOUS COMPACT GAS-COOLED REACTORS
Sawle, D. R. and Thomas, F. J. (Aerojet-General Nucleonics, San Ramon, Calif.
A/CONF.15/P/1884. 9p.

Various aspects of compact gas-cooled reactors are considered. The maximum conceivable accidents for various designs, as well as hazards from less severe accidents, and the probability of accident occurrence are discussed. (J.E.D.)

- 28 - 13 SAFETY ASPECTS OF THE CALDER HALL REACTOR IN THEORY AND EXPERIMENT
Brown, G., Kronberger, H., Leslie, F. M. and Moore, J. (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.) and Mummery, P. W. (Atomic Energy Research Establishment, Harwell, Berks, Eng.)
A/CONF.15/P/267. 39p.

The design specification, construction requirements, and inspection which ensure the soundness of the reactors at Calder Hall and similar power stations are discussed. The nuclear and thermal kinetics of these natural uranium gas-cooled graphite-moderated reactors are examined, and an account is given of relevant experiments, including those carried out on the Calder Hall reactors. This work takes into account the presence of plutonium and the consequent increase in moderator temperature coefficient with irradiation level. An outline is given of the results of compatibility studies and of the effect of irradiation on fuel elements, on graphite (with particular reference to stored energy), and on constructional steels. In the light of the information obtained on the nuclear and thermal characteristics of the reactor, a number of hypothetical fault conditions are examined. These include loss of reactivity control, blower failure, and loss of pressure. (auth)

28 - 14 SAFETY CONSIDERATIONS FOR GAS COOLED THERMAL REACTORS OF THE CALDER HALL TYPE

Farmer, F. R. and Fletcher, P. T. (United Kingdom Atomic Energy Authority, Risley, Lancs, Eng.); and Fry, T. M. (Atomic Energy Research Establishment, Harwell, Berks, Eng.).
A/CONF.15/P/2331. 10p.

The principles of reactor safety are considered, with particular reference to the Calder Hall type of gas-cooled thermal reactor. The need for codes of practice for engineering design in the atomic energy field is discussed, and the aims of such codes are defined. The limits of radiation exposure, both for occupational workers and the general public, are discussed. The principles of safety adopted in design to ensure that in normal operation the reactor is at all times operating within safe limits are considered, and the factors governing these limits are discussed. In assessing the safety of a reactor system, it is necessary to predict the cause and course of possible fault conditions and also to consider the possibilities of multiple failures.

28 - 15 SAFETY SYSTEM FOR THE BROOKHAVEN NATIONAL LABORATORY MEDICAL REACTOR Lones, W.

In Fifth Tripartite Instrumentation Conference, Brookhaven National Laboratory, October 22-26, 1956 Minutes. (TID-7543), p. 61-63. February 1958.

Sums up and monitors the period and power signals and stops the test operation when the combined signals reach a set value.

28 - 16 THE SPECIFICATION OF FAULT CONDITIONS FOR GAS-COOLED GRAPHITE-MODERATED REACTOR SYSTEMS

O'Neill, T. J.
G.E.C. Atomic Energy Rev. 1, 166-73(1958) March.

A catalog of possible faults that could occur in gas-cooled graphite-moderated reactors is presented; the fundamental causes and prevention of reactor accidents are discussed.
(W.D.M.)

28 - 17 SUMMARY HAZARDS REPORT ON ENRICHED FUEL ELEMENT LOADING FOR THE BNL RESEARCH REACTOR

Corngold, N.
(BNL-3146) Brookhaven National Lab., Upton, N.Y.
May 24, 1956. Decl. July 12, 1956. 47p.

Discussions are included on the Brookhaven Pile and changes proposed, the nuclear physics of the enriched reactor, controls, kinetic behavior, temperature effects, fuel elements, hazards in normal operation and some credible accidents. (M.H.R.)

- 28 - 18 SUPPLEMENT TO REPORT ON THE BROOKHAVEN NUCLEAR REACTOR PREPARED FOR REACTOR SAFEGUARD COMMITTEE OF THE ATOMIC ENERGY COMMISSION
Borst, L. B.
(BNL-18) (Suppl.) Brookhaven National Lab., Upton, N. Y.
August 30, 1948. Decl. September 21, 1956. 133p.

Factors affecting the operating safety of the Brookhaven Reactor on which data are presented include: a survey on the possibility of earthquakes; a survey on the procedure used for fabrication of the fuel cartridge; the performance of leak test equipment; procedures for the transfer of radioactive material from the canal and heat generation in the canal; procedures for handling liquid wastes; a survey of meteorological data and their relationship to pile operation; a survey of procedures for startup of the reactor; high-level operating procedures; and procedures for scheduled and emergency shutdowns.
(C.H.)

- 28 - 19 WINDSCALE--COMMITTEE'S REPORT

Nuclear Eng., Vol. 2, No. 21, p. 510-2, December 1957.

How accident at Windscale in England, Oct 1957 was due to application of excessive amount of nuclear heating during release of Wigner energy, which caused exceptional rise in temperature, leading to can failure, uranium oxidation and, finally fire; summary of document Cmd. 302 "Accident at Windscale No. 1 Pile on Oct 10 1957" published by H. M. Stationery Office.

- 28 - 20 WINDSCALE REPORT

Eng & Boiler House, Rev., Vol. 72, No. 12, p. 414-15; December 1957.
see also Steam Engr., Vol. 27, No. 314, p. 102, December 1957.

Notes on accident occurred on Oct 10, 1957, on No. 1 pile at Windscale (Cumberland) plutonium-producing plant of United Kingdom Atomic Energy Authority; summary of White Paper, released by committee of inquiry, giving account of accident, cause of failure and steps taken.

29. SHIELDING

29 - 1

ANALYSIS AND RECOMMENDED MODIFICATIONS FOR SHIELD STRUCTURE--BROOKHAVEN NATIONAL LABORATORY FOR THE H. K. FERGUSON CO., INC.

Lloyd, R. G. and Brooks, E. A.

(M-4512) Babcock and Wilcox Co., New York.

July 24, 1950. Decl. Dec. 8, 1955. 206p, Subcontract PRO-111 (S-26-146), Supplement No. 1.

An analysis was made of the shield structure at the Brookhaven Reactor and recommendations are presented for reinforcing and altering the design to raise the strength level and increase the service life of the structure. Drawings and supporting calculations for the recommendations are included. (C.H.)

29 - 2

THE ATTENUATION CHARACTERISTICS OF BROOKHAVEN CONCRETE

Pratt, W. W., Fox, M., Kouts, H. J. and Rand, A. C.

(BNL-145) Brookhaven National Lab., Upton, N. Y.

December 17, 1951. Decl. March 19, 1956. 32p.

A set of measurements was carried out to determine the attenuation characteristics of Brookhaven concrete for neutron and γ -ray radiation. The radiation was detected with brass-walled G-M γ -ray counters, BF_3 neutron counters, a paraffin-lined methane-filled fast neutron counter, Au and In foils, and photographic films. The results obtained using the radiation filtering through the graphite reflector of the pile were compared with the results obtained using the radiation emerging directly from a sheet of natural U exposed to thermal neutrons from the pile. An investigation was made of the effectiveness of a thermal shield consisting of a layer of Fe inserted between the source and the concrete shielding. The attenuation of the component of γ radiation emerging perpendicular to the shield face was compared with the attenuation of the total radiation emerging from the shield. (auth)

29 - 3

A CRITICAL REVIEW OF ORNL SHIELD MEASUREMENTS: NEUTRON ATTENUATION

Sleeper, H. P. Jr.

(ORNL-436) (Del.) Oak Ridge National Lab., Tenn.

December 21, 1949. Decl. with deletions February 21, 1957. 47p.

Contract W-7405-eng-26.

The purpose of this report is to summarize the neutron data for the most valuable tests of uniform shield where both the shield attenuation and composition are considered well known. Also the results of heavy metal differential neutron measurements are reported, even though the composition of the neutron analyzer is not well known. In these experiments, measurements were made in a uniform "analyzer" shield with and without a heavy metal front slab. The difference is an index of the fast neutron shielding characteristics of the metal. (auth)

- 29 - 4 DETERMINATION OF THERMAL STRESSES IN CONCRETE SHIELDS
Hannah, I. W.
G.E.C. Atomic Energy Rev., Vol. 1, No. 4, p. 209-16, September 1958.

Discussion of thermal stress in concrete radiation shields in relation to Hunterston type of reactor structure and its possible structural developments; factors involved in accurate determinations of thermal stresses; experimental studies.

- 29 - 5 THE DISTRIBUTION OF γ -RAYS AND NEUTRONS IN THE CONTROL FACE SHIELD OF B.E.P.O.
Harrison, J. R., Mills, A. M. and Bendell, D.
(AERE-RP/R-1604) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.
1955. 8p.

Measurements of neutron and γ -ray fluxes in the control face shield of B.E.P.O. have been made. The measurements extend over the entire thickness of the shield (6 ft 6 in. Barytes concrete), and cover a range of 10^7 in γ -ray and thermal neutron intensities. It is shown that 43 in. of concrete would have been sufficient for maximum permissible levels for all radiations to have been obtained.

- 29 - 6 PIPPA SHIELD DESIGN
Horton, C. C.
(TRDC/P-3) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.
July 9, 1953. 13p.

Previous calculations on the Pippa shield are revised in the light of fresh design data for Pippa and recent experimental data. The shield thickness of concrete, heat production in the shield, the activity of the pressure vessel, the activity of coolant and void air, and the activity of the asbestos linings are considered. Thermal neutron currents are calculated for the radial case, and for the axial case the observed datum

from Windscale is used: that the emergent axial current is 0.01 of the maximum thermal neutron flux at the center. Flux calculated by alternative methods are shown for comparison. (M.H.R.)

29 - 7

SHIELDING STUDIES. (PART) 1. FAST NEUTRON FLUXES FROM GRAPHITE MODERATED URANIUM REACTORS

Absalom, R. M., Caunter, D. R., Horton, C. C., Lea, M. U. and Shepherd, L. R.
(AERE-R/R-842) (Del.) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.
December 1951. Decl. with deletions November 26, 1956. 33p.

Experiments have been performed on the attenuation of fast neutrons from a U fission neutron source through thicknesses of graphite up to 1 meter, and through two different thicknesses of steel. The neutrons were detected by two different kinds of counter; a thick polythene radiator whose response is energy-dependent above a threshold of ~ 800 kev; and, a flat response counter (long counter) whose response is essentially independent of energy between at least 30 kev and 5 Mev. The space distribution of both groups of neutrons may be fitted to simple formulae. The distribution of 2-Mev neutrons in the core and reflector of BEPO has also been calculated; measurements have been made using the sulfur activation methods, which show reasonable agreement with the calculated values. The neutron energy spectrum above 1 Mev in BEPO has been measured using the recoil proton method in Ilford 100 μ C2 emulsions, these measurements were combined with the sulfur activation method mentioned above to obtain the fast neutron flux. The neutron spectrum above 1 Mev from thermal fission of U^{235} has also been measured, this measurement combined with the known energy dependence of the polythene detector indicates the effective energy of the neutrons measured by this detector to be 2 Mev. (auth)

29 - 8

A STUDY OF THE NUCLEAR AND PHYSICAL PROPERTIES OF THE ORNL GRAPHITE REACTOR SHIELD

Blosser, T. V., Bond, G. W., Lee, L. A., Morgan, D. T., Nichols, J. F., Reid, R. C., Reynolds, A. B., Speidel, T. O. P., Vroom, D. W. and Welt, M. A.
(ORNL-2195) Oak Ridge National Lab., Tenn.
September 8, 1958. 43p. Contract W-7405-eng-26.

A study of the nuclear and physical properties of the concrete shield of the ORNL Graphite Reactor was performed both to determine the radiation attenuation characteristics of the shield and to discover any effects of a long-term (12-year) irradiation. In the experiment neutron and gamma radiation measurements were made in a 4-5/8-in.-diam. hole in the shield as the

hole was drilled in increments of 1 or 1/2 ft. Concrete plugs were always placed behind the detectors to simulate a homogeneous shield. The average relaxation length for gamma rays in the barytes-haydite concrete, of which most of the shield is made, was found to be approximately 13.6 cm. increasing gradually from 13.0 cm at a shield thickness of 2 ft to 14.6 cm at a shield thickness of 5 ft. The fast-neutron relaxation length in the barytes-haydic concrete varied from 10.0 cm to 10.6 cm, the average being approximately 10.2 cm. Measurements of the streaming of radiation through the hole were also made. The concrete dust collected from the drillings was used to determine the chemical composition, water content, density, compressive strength, and radioactivity of the shield at the various depths. The temperature gradient through the shield was also measured. This investigation showed that the chemical properties and density of the shield have not changed appreciably since a similar investigation in 1948, but its compressive strength is lower (40% near the reflector-shield interface).
(auth)

30. SHIP PROPULSION

30 - 1 C.C.G.C. REACTOR FOR SHIPS

- - -

Atoms and Nuclear Energy 9, 162-3(1958) May.

Although the nuclear engines of U. S. submarines have proved the possibility of atomic power for ship propulsion, it is desirable to find quickly one or more cheaper and more efficient marine engines based on nuclear power. The U. S. is exploring every possible system. A study is being made of a closed-cycle gas-cooled reactor with the object of incorporation into a tanker by 1961. In spite of its many difficulties, the closed-cycle gas-cooled reactors offers the big advantage of high efficiency. (A.C.)

30 - 2 GAS COOLED NUCLEAR REACTOR STUDY. FINAL REPORT

Thompson, A. S.

(AECU-3559) Studebaker-Packard Corp. Nuclear Power Dept., Detroit.
July 31, 1956. 110p. Contract AT(30-3)-214.

An investigation was made of the performance of a gas-cooled reactor, designed to provide a source of high temperature heat to a stream of helium. This reactor, in turn, is used as a source of heat for the air stream in a gas-turbine power plant. The reactor design was predicted primarily on the requirement for transferring a large amount of heat to the helium stream with a pressure drop low enough that it will not represent a major loss of power in the power plant. The mass of uranium required for criticality under various circumstances was investigated by multigroup calculations, both on desk calculators and on an IBM-704 machine. The gas-turbine power plant performance was studied based on a Studebaker-Packard-designed gas-turbine power plant for the propulsion of destroyer-escort vessels. A small experimental program was carried out to study some effects of helium on graphite and on structural steels.
(auth)

30 - 3 KINETICS STUDY FOR CLOSED CYCLE GAS COOLED REACTOR PROGRAM

- - -

(FICO-108) Ford Instrument Co., Long Island City, N. Y.
March 1958. 64p. Contract AT(30-0)-250.

The transient behavior of the closed cycle, gas cooled reactor system for a 20,000 shaft horsepower merchant ship was investigated using a digital differential analyzer. In this cycle, the helium coolant is heated in the reactor and then passes through the high pressure turbine, a throttle valve, low pressure turbine provided with a bypass valve, the hot side of the recuperator, precooler, compressor, recuperator cold side, and back to the reactor inlet. Step function changes in reactor reactivity and heat release and independent perturbations of control elements at steady state full load conditions establish basic trends free from interaction effects. The basic requirement of constant reactor outlet temperature was postulated for reactor power control. This parameter was held fixed and the effects of ramp function closures of the throttle valve and openings of the bypass valve were evaluated. The system power level was changed by extracting and introducing gas at both the high and low pressure legs of the system. Finally, the study presents the results of two modes of system control for maneuvering operation. The demands of this condition were met by bypassing the low pressure turbine and positioning the throttle valve by (a) a regulatory signal from the high pressure turbine shaft speed and (b) programmed interlocks on bypass valve position. (auth)

30 - 4

NUCLEAR POWER FOR COMMERCIAL VESSELS

Maddocks, K. (Univ. of Michigan, Ann Arbor).

Atomics 7, 151-5, 166(May); 197-201, 224(June); 291-5(August); 309-14(September) 1956.

A survey is presented of British and American unclassified material relative to the use of nuclear power for marine propulsion. Following a brief discussion on the principles of fission and reactor operation, five types of reactors suitable for marine use and one type suitable for fuel production are described and illustrated. The gas cooled reactor is selected as most suitable for marine propulsion and a proposed closed cycle gas turbine plant is analyzed in some detail. Various proposals for the use of nuclear power in specific ships are reviewed, and an economic analysis is made to compare a 30,000 ton d.w. tanker when operating with an oil fired steam turbine plant and when operating with a nuclear-powered closed-cycle helium turbine. (auth)

30 - 5

NUCLEAR-POWERED SHIPS FOR AMERICAN SHIP OPERATORS

(TID-7539) Division of Reactor Development. Maritime Reactors Branch, AEC and Maritime Administration, Washington, D. C.
July 30, 1957. 196p. A Symposium Held at Washington, D. C.

Fifteen papers presented at the Washington symposium are included. Marine applications of nuclear power are discussed with regard to maritime regulations and marine insurance. The designs of the first nuclear merchant vessel and its pressurized water reactor are given. Other reactor designs proposed for merchant ship applications are boiling water, organic moderated, and gas cooled reactors. (D.E.B.)

30 - 6

THE NUCLEAR PROPULSION NO. OF SHIPS

Hinton, C. and Moore, R. V. (United Kingdom Atomic Energy Authority).
J. Joint Panel Nuclear Marine Propulsion 2, 1-25(1958) April.

After briefly reviewing the historical background of power driven ships and inferring some analogous future trends, the characteristics of pressurized water and gas cooled, graphite moderated reactors are reviewed in general terms. A gas cooled, graphite moderated reactor of high rating designed for the propulsion of a super-tanker is described. The economics of operation of a nuclear powered supertanker is compared with one which is conventionally powered. Some of the practical problems, including control and safety, of a nuclear powered ship are defined and discussed. A review of some reactor types having potential merit for the marine application is included. While nuclear power units of 20,000 to 25,000 shp with competitive economic performance could be designed in the next few years, a reactor suitable for powers below these values has yet to be developed. (auth)

30 - 7

POSSIBILITIES AND CONDITIONS OF USE OF WATER OR GAS TYPE REACTORS IN A TANKER. I. PRESSURIZED WATER REACTOR

L. Bureau du Colombier, Hainzelin, J. and Duquesne, P. (Chantiers de l'Atlantiques, Paris). II. BOILING WATER REACTOR Ricard, J. and Domenjoud, P. (Société des Forges et Ateliers du Creusot, Paris). III. GAS REACTOR. Gibrat, R. (Société Indatom, France).
 A/CONF.15/P/1137. 79p.

The French Atomic Energy Commission and the Secretary of State for the Merchant Navy have organized a study for the design of a nuclear-propelled oil tanker of 40,000 tons and 20,000 hp maximum power. Three types of reactors were considered: gas cooled, pressurized water, and boiling water. This report, issued from the study, shows the characteristics and difficulties peculiar to each of these solutions and describes a number of particular dispositions permitting the installation of such reactors on board ship. (auth)

30 - 8 REVIEW OF ATOMIC PROPULSION

Brard

Bull. Tech. Bur. Veritas 40, 179-92(1958) April. (In French)

The fundamentals of the fission process and the suitability of the various possible reactor fuels are reviewed. The question of the critical size of a reactor and possible types of reactor for marine propulsion are considered. Fuel utilization and methods for prolonging the life of the fuel rods are discussed. Some details are given on the operation of a reactor. Pressurized water reactors, boiling water reactors, and gas cooled reactors are described in detail, and the subject of shielding is discussed. Other subjects considered include the design of the reactor pressure vessel, piping, and evaporators, and the weight, space required, and costs for the nuclear installation. (auth)

30 - 9 SHIP PROPULSION WITH ENCLOSED GAS TURBINES AND GAS COOLED REACTORS

Bammert, K.

Atomkern Energie 2, 381-4(1957) October. (In German)

Describes the newest project of a ship propulsion system using N_2 as coolant for the high temperature closed cycle reactor. (F.S.)

30 - 10 SHIPS: GAS-COOLED REACTOR IS THE CHOICE

Hinton, C. (Central Electricity Generating Board) and Moore, R. V. (U. K. Atomic Energy Authority).

Nuclear Power 3, 11-16(1958) January.

The construction of a nuclear power ship by Great Britain necessitates the choice of reactor type. A review of the economic factors and other problems leads to the conclusion that a gas-cooled reactor of the Calder Hall type is the most feasible. (J.S.R.)

31. SITE PREPARATION

- 31 - 1 OAK RIDGE SITE REPORT, 30,000 KW PROTOTYPE PARTIALLY ENRICHED URANIUM GAS COOLED, GRAPHITE MODERATED NUCLEAR POWER PLANT (IDO-24028) Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif. and Nuclear Products-ERCO, Div., ACF Industries, Inc. Washington, D. C. March 1959. 65p. Contract AT(10-1)-925. Report No. 59-4-RE.

The PEGCPR is a partially enriched uranium fueled, helium cooled, nuclear power plant to be constructed at the National Reactor Testing Station, Idaho. Design modifications were included to construct the PEGCPR at Oak Ridge. This report describes the selected site, the reactor safety aspects for the site and the modified design of the prototype plant. Cost estimates are included.

- 31 - 2 SAVANNAH RIVER PROJECT SITE REPORT 30,000 KW PROTOTYPE PARTIALLY ENRICHED URANIUM GAS-COOLED GRAPHITE MODERATED NUCLEAR POWER PLANT - - - (IDO-24029) Kaiser Engineers Div., Henry J. Kaiser Co., Oakland, Calif. and Nuclear Products - ERCO Div., ACF Industries Inc., Washington, D. C. March 1959. 60p. Contract AT(10-1)-925. Report 59-5-RE.

This report is a feasibility study and cost estimate per a modified PEGCPR plant (originally designed for the National Reactor Testing Station, Idaho) modified and adapted to a site at the Savannah River Project, Atken, SC. Modifications of this prototype partially enriched uranium fueled, helium cooled, nuclear power plant are mostly in the design of the buildings, structures, utilities and service facilities. The basic design of the reactor is unchanged.

- 31 - 3 SITE ORGANIZATION AT CALDER HALL Varcoe, J. F. (Taylor Woodrow Construction, Ltd.) Nuclear Eng. 2, 135-8(1957) April.

The planning and organization carried out at the Calder Hall construction site is described. The handling of labor, materials, security, and coordination is discussed. (T.R.H.)

32. STARTING

32 - 1 AUTOMATIC START-UP OF NUCLEAR REACTORS

Cox, R. C.

Inst Radio Engrs--Trans on Nuclear Science, Vol. NS-3, No. 1, p. 15-20.
February 1956.

Bringing reactor from subcritical state to neutron power level that produces significant temperature rise is operation that can only be performed by controlling reactor on neutron flux information; this operation can be performed automatically; details of equipment installed on British experimental pile BEPO; block schematic diagrams showing arrangement of servos, amplifiers and period meter.

32 - 2 COMMISSIONING OF CALDER HALL NO. 1 REACTOR

Brown, G., Stretch, K. L.

Engineer Vol. 202, No. 5261, 5262 November 23, 1956. p. 722-6,
November 30, p. 760-3.

Commissioning and start-up procedure; proving tests on plant prior to hand-over to operational team responsible for actual start-up, including loading of uranium; loading to critical size and to full size; air flow and CO tests.

33. THERMODYNAMICS

- 33 - 1 CYCLING SCHEMES FOR THE GAS-COOLED HIGH-TEMPERATURE REACTOR
Spillmann, W. and Tognoni, R. (Escher Wyss AG, Zurich).
Atomkern Energie 4, 190-95(1959) May. (In German)

For gas-cooled reactors two thermodynamic cycles working according to different principles are used namely the direct single loop and the two circuit system with indirect heat addition to the working medium. While no losses and no external pumping power are needed for the single loop cycle, various possibilities are discussed for the two loop cycle. A table giving characteristics of He, N₂, Ne, CO₂ and air as coolants is also included.

- 33 - 2 DEPENDENCE OF PERFORMANCE OF GAS COOLED NUCLEAR REACTORS ON THE COOLING GAS PRESSURE AND THE REACTOR DIMENSIONS
Schraud, A.
Atomkern Energie 2, 204-7(1957) June. (In German)

Since the British atomic power program has given prominence to gas cooled reactors, relationships are discussed under the following headings: connection between effective and heat performance, dependence of heat performance, degree of cooling effect, heat transfer and pressure drop in the central channel, dependence of reactor performance.

- 32 - 3 THE DERIVATION OF REACTOR HEAT TRANSFER TRANSIENT EQUATIONS FOR GAS COOLED GRAPHITE MODERATED THERMAL REACTORS
O'Neill, T. J. (General Electric Co., Ltd., Erith, Kent, Eng.).
A/CONF.15/P/21. 28p.

In making transient studies on the behavior of gas-cooled graphite moderated thermal reactors, one of the sets of mathematical equations involved is that for reactor heat transfer. These equations have for inputs: reactor power reactor gas flow and reactor inlet gas temperature; and produce as output; reactor outlet gas temperature, the uranium and graphite reactivity affecting variables, and the maximum fuel element temperatures. From the consideration of a unit length of one fuel element channel in the core, partial differential equations of the system are determined. These are integrated in space,

to give the form required for analogue computer studies, i.e., simultaneous ordinary nonlinear differential equations. The final equations are given for two variants in moderator design, viz: a solid block type of moderator, and a moderator involving in part a graphite fuel element supporting sleeve. Account is taken of heat transfer by conduction in the solids, and by convection and thermal radiation in and between the gas spaces.

33 - 4

EFFECT OF GRAPHITE TEMPERATURE ON POWER SENT OUT BY PIPPA

Bowen, J. H.

(NURG-M/59) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England.

May 12, 1953. 12p.

A preliminary examination is made of the variation of electrical power output of Pippa type plants, with straight through and re-entrant cooling, over a range of mean graphite temperatures. (auth).

33 - 5

A GAS-COOLED NUCLEAR REACTOR OF HIGH THERMAL EFFICIENCY

Winterberg, F.

Atomkern Energie, 4, 2-6, (1959) January. (In German)

This report shows that the thermal efficiency of a gas-cooled reactor with gas-turbines can be increased by isothermal expansion of the coolant in appropriate cooling channels of variable cross section in the reactor core. To optimize the efficiency of the cooling, it is desirable to hold the surface temperature of the cooling channels everywhere to a constant. Under this condition the cooling-channel cross section, the flow velocity and the heat source distribution of the coolant are calculated.

33 - 6

A MONTE CARLO STUDY OF THERMAL UTILIZATION FACTOR AND DIFFUSION AREA: GAS-COOLED GRAPHITE-MODERATED LATTICES

Schaefer, G. W. and Parkyn, D. M. (English Electric Co., Ltd.. Whetstone, Leics, Eng.).

A/CONF.15/P/310. 16p.

An attempt is made to explain the results from thermal flux in gas-cooled graphite-moderated uranium lattices experiments by solving the one-energy-group transport equation by the Monte Carlo method, using fundamental cross sections. The experiments of Campbell et al. and the theory of Amouyal et al. are compared with the Monte Carlo results. A survey of diffusion areas and asymmetries was carried out, using a number of mathematical models. The results are compared with theory

and experiment. The survey is restricted to cylindrical fuel rods in circular channels and square pitches. (W.D.M.)

- 33 - 7 THE POSSIBILITIES OF ACHIEVING HIGH TEMPERATURES IN A GAS COOLED REACTOR
Shepherd, L. R., Huddle, R. A. V., Husain, L. A., Lockett, G. E.,
Sterry, F. and Wordsworth, D. V. (Atomic Energy Research Establish-
ment, Harwell, Berks, Eng.).
A/CONF.15/P/314. 36p.

Large natural uranium gas cooled power reactors have limitations with regard to core heat output density, which implies that they must be relatively cumbersome. They are also limited in maximum attainable coolant outlet temperature. There is an obvious incentive to develop compact reactors having core power densities as high as pressurized water reactors, but not possessing the coolant temperature limitations either of pressurized water reactors or of the more advanced natural uranium gas cooled systems. Coolant outlet temperatures in excess of 650°C are needed in order to exploit the most modern steam turbine generating equipment. A higher outlet temperature of 750°C is desirable for gas turbine application or for a mixed steam/gas turbine system. Some thought is being given to the possibility of a high temperature gas cooled reactor in which an enriched fuel is incorporated in a solid moderator to give fuel elements of greatly extended heat transfer surface area, thereby permitting a high density of heat output from the core. The core materials are in a highly refractory form (e.g., graphite and ceramics) and metal fuel element sheaths are avoided. These materials will be allowed to run to much greater temperatures than have been possible in metal fuel elements and, as a result, high gas temperatures can be obtained. (auth)

- 33 - 8 THE SACLAY REACTOR: TWO YEARS OF EXPERIENCE IN THE USE OF A COMPRESSED GAS AS A HEAT TRANSFER AGENT
Yvon, J.
In International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Proceedings. Vol. 2, Physics: Research Reactors. New York, United Nations, 1956. p. 337-350.

The design history of the Saclay reactor is outlined and the reactor is described. The description includes the pressure vessel, fuel assemblies, reflector and shielding, controls, experimental facilities, heavy water circuit, recombination circuit, compressed gas circuit, water circuit, and air circuit. The power level was computed from the neutron flux and from the thermal power and found to be 11% greater when calculated from the neutron flux. A study was made of the reactivity components and the long term reactivity loss was computed. The heat transfer techniques in use in reactors are discussed and the factors affecting the choice of a gas are listed. The

nuclear and thermal requirements, speed of the gas, output and temperature distribution, pressure drops, and leaks are discussed. The discussion concludes with the prospects offered by the use of compressed gases as reactor coolants.

33 - 9

TEMPERATURE TRANSIENTS IN GAS-COOLED THERMAL NUCLEAR REACTORS
Bowen, J. H. and Masters, E. F. O. (U.K.A.E.A., Industrial Group H. Q., Risley, Lancs, Eng.).

J. Brit. Nuclear Energy Conf. 3, 188-200(1958) July. also
Instn. Elec. Engrs. Proc. Vol. 105, pt. B, No. 22, July 1958.
p. 337-49.

The transient behavior of the Calder Hall type reactor in terms of the design and operating parameters is examined, including the transients arising from step changes in reactivity, steady changes in reactivity, step changes in cooling, and steady changes in cooling due to both circular-speed and gas-pressure changes. The program was carried out by means of an analog computer and by a graphical method which employs an approximate solution of the neutron-kinetic equations. It is concluded that the graphical method is useful for studying the transient characteristics of a reactor; however, experimental confirmation of the predicted results is desirable. (J.R.D.)

33 - 10

TEMPERATURE TRANSIENTS IN PIPPA DURING THE INITIAL STAGES OF DIVERGENCE

Wordsworth, D. V.

(NURG/M-48) Gt. Brit. Atomic Energy Research Establishment, Harwell, Berks, England. January 9, 1953. 28p.

Uranium, coolant, and graphite temperatures have been calculated for PIPPA assuming an instantaneous rise in heat production. The times taken for these component temperatures to reach 95% of their new equilibrium values have been compared with the results of similar calculations on the Harwell Pile. It is found that temperature changes will be much more rapid in PIPPA, the U taking 18 mins compared with 54 mins in the Harwell Pile, the coolant 26 mins compared with 2.4 hrs and the graphite 53 mins compared with 3.2 hrs. Further expressions have been found for the component temperatures in PIPPA following an instantaneous change in reactivity Δk , and allowing for the delayed neutron groups. It is found that to give good estimates of temperature in the first minute of the divergence it is necessary to take into account all the terms in the solution. After that time satisfactory results would be given if only the dominant

term was considered, but it must be noted that the method can only give good estimates in the early stages of divergence since the effects of temperature on reactivity are not included. (auth)

33 - 11 THERMAL CAPACITY IN REACTORS

Ledinegg, M.

Atomkern Energie 4, 14-22(1959) January (In German)

The object of the study was to determine the highest thermal capacity of reactors. One of the most important limits for gas and liquid cooled reactors is the highest temperature inside the fuel elements. Some other factors are considered. A relation for the highest thermal capacity is deduced, and the best possible values for the dimensions with regard to minimum work of the circulation pumps are accounted. Other possibilities for its reduction, like radial and axial variation of the cross section of the cooling channels, are thoroughly examined. (auth)

33 - 12 A THERMAL CYCLE FOR NUCLEAR POWER PLANT WITH GAS COOLED REACTOR

Juza, Jan. (V. I. Lenin's Plant, Pilsen).

Jaderna Energie 4, No. 1, 2-9(1958) January. (In Czech.)

The effects of the cooling gas temperature at the reactor inlet, reheating of vapor from the low pressure turbine, and preheating the feed water on the efficiency of one or two stage vapor cycles generated in a steam generator and on the circulating gas blower input were evaluated. The influence of the parameter changes, the costs of primary cycle, and the breeding of nuclear fuel were not considered. The results showed that the suggested thermal cycle with two pressure stages (70 to 160 atm. and 15 to 25 atm.) with preheating of the feed water and reheating of vapor affected the gas temperature of 175 to 250°C at the inlet and of 400 to 500°C at the outlet of the reactor. The thermal efficiency of the power plant was 27.6 to 32.6% (tr-auth)

33 - 13 THERMO-DYNAMIC CALCULATION OF GAS-COOLED REACTORS

Baehr, H. D.

Atomkern Energie, 4, 7-11, January. 49-55, February(1959)
(In German).

The equations governing the relations between power, gas temperatures, pressure drop, and pumping power for a single

cooling channel are presented graphically in a diagram of channel characteristics. By means of this diagram various possibilities for the gas distribution to each of the cooling channels can be examined. The economic optimum of the pumping power can be found by trial and error with the aid of the channel characteristics. An analytical solution was found for a special coolant distribution. Finally the example of a reactor cooled by high-pressure gas is presented.

Part II discusses the heat removal from the reactor, particularly the distribution of coolant flow, pumping achievement and reactor heat production, and the production of electricity.

- 33 - 14 THERMODYNAMIC STUDY OF THE G-2 AND G-3
Chaboseau, J. and Laferriere, P.
Bull. inform. sci. et tech. No. 20, 10-27(1958) August
(In French)

A thermodynamic study of the G-2 and G-3 reactors must consider the plutonium generation function of the reactors and the recovery of thermal energy in a power plant with as good a yield as possible. The formulas used for the calculation of heat transfer and charge loss are derived. The conditions for the start-up of the reactors are determined. A study was made of the cooling of the reactors in the case of the flattening of the radial flux curve. (J.S.R.)

- 33 - 15 THERMODYNAMICS OF GAS-COOLED REACTORS
Melese, G. B.
Nucleonics 16, No. 2, 72-6(1958) February.

A thermodynamic analysis of gas-cooled reactors in terms of simplified basic equations was made. The resulting expression for electrical power output can be written as a product of five factors: (1) outside variables, (2) nuclear quantities, (3) maximum can temperature, (4) dimensions and gas pressure, and (5) properties of the coolant gas. (J.S.R.)

34. WASTE DISPOSAL

34 - 1

THE ORIGIN AND DISPOSAL OF POWER REACTOR WASTES

Dix, G. P., Groscup, R. C. and Leffler, J. M.

(MND-1235) Martin Co. Nuclear Div., Baltimore

February 5, 1958. 104p.

A fundamental investigation was made of waste disposal pertinent to pressurized water power reactor types. A study of gas-cooled reactor types was included to determine the specific waste disposal aspects concerning a maritime gas-cooled reactor. The study included a thorough search of the current literature. The study encompasses only operational power reactor wastes and excludes the disposal of used fuel elements. The properties of light water, air, nitrogen, hydrogen, helium, and carbon dioxide were studied relative to their use as reactor coolants. The origin of reactor wastes is discussed at length. The waste collection components of a power reactor system are described in detail. Various methods of disposing of reactor wastes are outlined. A distinction is made between activation and fission products. A list is tabulated of radioisotopes produced in coolants by activation during normal reactor operation. Design details are described of waste disposal systems of five operating or proposed power reactors. These include the Shippingport Pressurized Water, Army Package Power, Martin Pressurized Water, Maritime Gas Cycle, and the Calder Hall Reactors. The shielding of waste disposal systems is discussed. Methods of calculating radiation doses from reactor waste collection components of pressurized water reactors are outlined. Codes and standards are outlined which are applicable to waste disposal as set forth by the U. S. Atomic Energy Commission, Interstate Commerce Commission, and U. S. Coast Guard. A list of the permissible concentrations of radioisotopes in air and water is included. (C.H.)

GAS-COOLED REACTOR DIRECTORY

Reactors are classified by moderator and listed in chronological order of criticality. They are shown as either in operation (O), under construction (C), or definitely planned (P). Power reactors: output is from one reactor unless otherwise stated in brackets; coolant temperatures are mainly average in the core unless given as exit values. Research reactor neutron flux figures are in n/cm³/sec.

POWER REACTORS

Type	Name and/or owner	Status and first loading	Location	Contractor	Net output MWe	Fuel type(enrichment) form, cladding	Coolant type, pressure(psia) temp(°C), flow(lb/min)	Notes
GRAPHITE	CALDER HALL A&B (UKAEA)	O 56,58	Sellafield, England		140(4)	Nat U,rod,Magnox	CO ₂ ,115, 333(ex), 33,300	All critical, Pu prod
	CHAPELCROSS A&B (UKAEA)	O,C 58,59	Annan, Scotland		140(4)	Nat U,rod,Magnox	CO ₂ ,115, 333(ex), 33,300	1 critical, Pu prod
	G-2, G-3 (CEA)	O,C 58,59	Marcoule, France		60(2)	Nat U,rod,Mg-Zr alloy	CO ₂ ,210, 309(av), 100,000	
	EDF-1 (Electricite de France)	C mid 59	Chinon, France		60	Nat U,rod,Mg alloy	CO ₂ ,350,355(ex), 167,000	Pu prod
	EDF-2 (Electricite de France)	C 61	Chinon, France		167	Nat U,rod,Mg alloy	CO ₂ ,350,355(ex), 167,000	No containment vessel
	BRADWELL (CEGB)	C early 61	Bradwell, England	Nuclear Power Plant Co.	300(2)	Nat U,rod,Magnox	CO ₂ ,147,390(ex)	
	AGR-Advanced Gas-Cooled Reactor(UKAEA)	C early 61	Windscale, England		28	UO ₂ (1.25%), pellets, Be	CO ₂ ,285,575(ex)	
	BERKELEY (CEGB)	C mid 61	Berkeley, England	AEI-John Thompson	275(2)	Nat U,rod,Magnox	CO ₂ ,140,345(ex), 381,000	
	HUNTERSTON (SSEB)	C June 62	Ardrossan, Scotland	GEC/Simon-Carves	300(2)	Nat U,rod,Magnox	CO ₂ ,165,391(ex), 354,000	
	HINKLEY POINT (CEGB)	C end 62	Bridgwater, England	English Electric Group	500(2)	Nat U,rod,Magnox	CO ₂ ,200,375(ex), 600,000	
	LATINA power station (SIMEA)	C 62	Latina, Italy	NPFC/AGIP Nuc	200	Nat U,rod,Magnox	CO ₂ ,390	
	ML-1 Mobile Low Power Plant No. 1 (AEC)	P 61	NRTS, Idaho	Aerojet-General	400		Gas	
HEAVY WATER	Philadelphia Electric Co	P 63	Peach Bottom, Pa.	General Dynamics	28.5		Gas	
	EAST CENTRAL AND FLORIDA WEST COAST Nuclear power Groups	P 63	Florida	GNE	46	U(slight)	Gas	High temp pressure tube prototype for 200 MW station
	CZECH Station	P 63	Bohunice, Czechoslovakia		150	Nat U, Be-Mg	CO ₂ ,850,425	

GAS-COOLED REACTOR DIRECTORY (Continued)

POWER REACTOR EXPERIMENTS AND DESIGN STUDIES

Type	Name and/or owner	Status and first loading	Location	Contractor	Net output MWe	Fuel type(enrichment) form, cladding	Coolant type, pressure(psia) temp(°C), flow (lb/min)	Notes
GRAPHITE	G-1 (CEA)	O 56	Marcoule, France		5e	Nat U, Mg	Air, atm, 220, 12,700	
	GCRE high temp gas-cooled exp. (AEC)	C 59	NRTS, Idaho	Aerojet-General			N 540	
	GTTF-gas turbine testing facility(Army)	C	Fort Belvoir, Va.	Aerojet-General			212, 620	
	PEGPR (AEC)	P 62	Oak Ridge, Tenn.	Kaiser Engineers-ACF	30e	UO ₂ (2%)	He, 387,540	Prototype
	MGCR (AEC & Maritime Admin)	P 63	San Diego, Calif.	General Dynamics	80t	UO ₂ ,stainless steel	He 700	Prototype, shaft horsepower 20,000
	HTGR-High Temp. Gas-Cooled (UKAEA-ENEA)	P	Winfrith Heath, England		10t		He	
	TURRET (AEC)	P 61	Los Alamos, N. M.	LASL	3t	U(slight),cylinder unclad	N 700	Closed cycle gas turbine power plant
	GCR-2 (AEC)		Oak Ridge, Tenn.	ORNL	225e	UO ₂ ,(slight), stainless steel	He 540	Design study
	DANIELS Experimental Power Pile (AEC)		Oak Ridge, Tenn.	ORNL, Clinton Lab.	40t	U(slight)	He 650, 20,300	Design study
	GCPBR-ceramic power breeder (AEC)		Oak Ridge, Tenn.	ORNL		SiC-Si		Design study

GAS-COOLED REACTOR DIRECTORY (Continued)

RESEARCH REACTORS

Type	Name and/or owner	Status & first loading	Location	Contractor	Thermal output kW	Max neutron flux	Fuel type(enrichment) form, cladding	Reflector	Coolant
GRAPHITE	X-10 (AEC-ORNL)	O 43	Oak Ridge, Tenn.	Clinton Lab.	3800	1.12×10^{12}	Nat U, rod, Al	Graphite	Air, once thru
	GLEEP-Graphite Low Energy Experimental Pile (UKAEA)	O 47	Harwell, England		10	3.7×10^{10}	Nat U bars & UO_2 pellets, Al	Graphite	Air, once thru
	BEPO (UKAEA)	O 48	Harwell, England		6	10^{12}	Nat U bars, Al	Graphite	Air, once thru
	BROCKHAVEN Research Reactor (AEC)	O 50	Upton, N. Y.	H. K. Ferguson	30	3×10^{13}	U/Al alloy plates (93%)Al	Graphite	Air, once thru
	WINDSCALE 182 (UKAEA)	O 50,51	Sellafield, England				Nat U bars, Al	Graphite	Air, once thru
	BR-1 (CEAEN)	O 56	Mol, Belgium		4000	2.1×10^{12}	Nat U, bars, Al	Graphite	Air
	TREAT-Transient Reactor Test (AEC)	O 59	NRTS, Idaho	ANL	10^6	10^{16}	U, block, Zircalloy	Graphite	Air
	ZENITH-High Temp Zero Energy Reactor (UKAEA)	C 59	Winfrith Heath, Eng.	GEC		10^8	U,Th (slight), disks	Graphite	N
HEAVY WATER	EL-2 (CEA)	O 52	Saclay, France		2000	10^{13}	Nat U	Graphite	CO_2 at 7 atm
	R-1 (AB Atomenergi)	O 54	Stockholm, Sweden		300	10^{12}	Nat U, rods	Graphite	Air

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