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LOS ALAMOS SCIENTIFIC LABORATORY

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University of California

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**Coated Particle Fuels--Thermal, Mechanical,
and Stress Properties: A Bibliography**

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Compiled by

Jean Furnish

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COATED PARTICLE FUELS - THERMAL, MECHANICAL AND STRESS PROPERTIES
(FUELS LIMITED TO URANIUM COMPOUNDS COATED
WITH SOME FORM OF CARBON COMPOUNDS):
A BIBLIOGRAPHY

INTRODUCTION

The Nuclear Science Abstracts from 1964-1967 were surveyed to provide a background source of reference for the newcomer to the field of coated particle technology. Interest was concentrated in the area of uranium as the fissile material and directed toward mechanical and thermal properties of concern in stress analysis of the thermal stresses in such fuel particles and the interaction of such particles with the fuel particle having matrix.

The literature review is an extension of ORNL-TM-1335 which contains a machine-assembled review of coated nuclear fuel particles up to December 1965. In addition, references to radiation stability were included when thermal effects were mentioned.

The abstracts are arranged in chronological order by year, with the most recent given first. Four items located after the main body of the search was completed have been added at the end.

1.

13241 (ORNL-TM-1649) PYROLYTIC CARBON DEPOSITED FROM PROPANE IN A FLUIDIZED BED. Beatty, Ronald Lee (Oak Ridge National Lab., Tenn.). Jan. 1967. Contract W-7405-eng-26. 124p. Dep. mn. CFSTI \$3.00 cy, \$0.65 mn.

Thesis. Submitted to Tennessee Univ., [Knoxville].

A study was made of the influence of process variables on density, degree of preferred orientation, crystallite size, and deposition rate and efficiency of pyrolytic carbon coatings deposited in fluidized beds. The process variables employed were deposition temperatures from 1200 to 1900°C and propane supply rates from 0.05 to 3.2 cc per min per cm² of surface area being coated. An additional variable, total gas flow rate, was examined at deposition temperatures of 1300 and 1700°C at a propane supply rate of 0.2 cc per minute per cm². In addition to the fluidized substrates (450-micron thorium particles and three-sixteenths-inch diameter graphite disks) used, a stationary axial probe was immersed in the fluidized bed and coated in each experiment. The maximum coating density (2.1 g per cc) was obtained at a deposition temperature of 1200°C. The density decreased to a minimum of about 1.45 g per cc at 1500°C and increased again with increasing deposition temperature. High propane supply rates yielded isotropic coatings at all deposition temperatures. The coating deposition rate increased with higher temperature and with higher propane supply rate. Deposits formed on the stationary axial probes showed that at low temperatures the coating rate is fairly uniform along the axis of the fluidized bed. At higher temperatures most of the coating is deposited in the lower part of the fluidized bed. The most promising set of coating conditions found was the lowest deposition temperature examined, 1200°C, and the highest propane supply rate used, 3.2 cubic centimeters per minute per square centimeter. The coating produced under these conditions had a density of 2.0 grams per cubic centimeter and an anisotropy factor of one. The deposition rate was approximately ten microns per minute. (J.H.M.)

2.

10230 PYROLYTIC CARBON COATED PARTICLES FOR NUCLEAR APPLICATIONS. Bokros, Jack C.; Goedel, Walter V.; Chin, Jack; Price, Robert J. (to U. S. Atomic Energy Commission). U. S. Patent 3,298,921, Jan. 17, 1967.

Preparation of nuclear fuel particles coated with dense continuous jacket of isotropic C to increase stability under prolonged exposure to high temperature and neutron irradiation is described. (AEC)

3.

18354 (ORNL-TM-1653) IRRADIATION OF FUELED GRAPHITE FOR ADVANCED HIGH-TEMPERATURE GAS-COOLED REACTORS. Conlin, J. A.; Cunningham, C. W. (Oak Ridge National Lab., Tenn.). Mar. 1967. Contract W-7405-eng-26. 60p. Dep. mn. CFSTI \$3.00 cy, \$0.65 mn.

A series of six capsule-irradiation tests was made on cylindrical matrix-type fuel elements containing uncoated UC₂ particles or pyrolytic-carbon-coated UC₂ or (U-Th)C₂ particles. Those that contained uncoated fuel were encapsulated in low-permeability graphite as a barrier to fission-gas release. Fuel temperatures ranged from 2000 to 3500°F; power densities ranged from 10 to 300 W/cc; and burnups were up to 21 at. % ²³⁵U. One capsule contained loose pyrolytic-carbon-coated fuel particles in sealed tubes, in addition to a matrix element. Fission-gas release was determined during irradiation by sampling the helium purge gas for analysis by gamma-ray spectrometry. Information on the overall stability of the fuels was determined by postirradiation examination. Information is also included on the design of the experimental equipment, operation, techniques, and fission-gas release. (auth)

4.

16663 VALUE OF A SILICON CARBIDE BARRIER IN HTR FUELS. de Nordwall, H. J.; Sayers, J. B. (Atomic Energy Research Establishment, Harwell, Eng.). 31p. (CONF-661003-15). DTIE.

From AIME Nuclear Metallurgy Symposium on High Temperature Nuclear Fuels, Delavan, Wis.

Results of experiments to determine the performance of SiC as a fission product retaining barrier in reactor fuels are reported. Comparison is made between fission product retention of SiC coated particles and pyrocarbon coatings. Techniques for coating fuel particles with SiC and for fabrication of SiC-matrix fuels are discussed. Other topics covered include fission product release at very low burn-up, metallurgical examination of high burn-up particles, migration of fissile material, and fission product retention in high-temperature reactor fuel elements. (P.G.C.)

5.

18266 (ORNL-TM-1713) PREIRRADIATION EVALUATION OF THE PYROLYTIC-CARBON-COATED FUEL AND THE ASSEMBLY FOR THE ORR-A9-4 CAPSULE. Hewette II, D. M.; Coobs, J. H.; Beatty, R. L.; Kegley, T. M.; Leslie, B. C. (Oak Ridge National Lab., Tenn.). Feb. 1967. Contract W-7405-eng-26. 50p. Dep. mn. CFSTI \$3.00 cy, \$0.65 mn.

Results are presented for preirradiation evaluation of pyrolytic-carbon-coated UO₂ and UC₂ particles and preparation of the fuel assembly for the ORR-A9-4 capsule. In the metallographic evaluation of the fuel particles, it was discovered that the microstructure observed for a given type of coating was dependent on the polishing technique. (auth)

6.

20591 (CONF-660511-, pp 1.7.1-16) TECHNIQUES FOR IRRADIATING HIGH TEMPERATURE MATERIALS IN A STEEP FLUX GRADIENT. Kellhotz, G. W.; Moore, R. E.; Osborne, M. F.; Wieland, B. W.; Zulliger, A. F. (Oak Ridge National Lab., Tenn.).

The J-12 core position of the Engineering Test Reactor is used to irradiate high-temperature materials under a variety of temperature and high neutron flux conditions. Specimen types include ceramic oxides, refractory metal carbides and nitrides, tensile test specimens, and coated fuel particles. A maximum fast neutron dose of 6×10^{21} neutrons/cm² can be achieved in one calendar year. The techniques evolved from design and development based on operating experience and the need to accommodate new specimen materials and irradiation condition requirements. Several multiple-capsule assemblies and the facility in which they are operated are described. (auth)

7.

21919 (ORNL-4076, pp 81-115) GAS-COOLED REACTORS. (Oak Ridge National Lab., Tenn.).

Research progress is reported on: diffusion processes, behavior of graphite with reactive gases, irradiation behavior of high-temperature fuel materials, and behavior of high-temperature materials under irradiation. (M.C.G.)

8.

20554 DISPERSION AND BREAKAGE OF COATED PARTICLES CONSOLIDATED IN A GRAPHITE MATRIX BY HOT PRESSING. Williams, G. H.; Morris, J. B.; Savage, D. W. (Atomic Energy Research Establishment, Harwell, Eng.). Proc. Brit. Ceram. Soc., No. 7, 41-60 (Feb. 1967).

One type of fuel element suitable for use in a high-temperature gas-cooled nuclear reactor consists essentially of compacts con-

taining a dispersion of coated-fuel particles embedded in a graphite matrix. For producing this type of compact a rapid hot-pressing process was developed. The effects of various parameters on the particle dispersion were examined. The most regular dispersion was achieved by the overcoating method of mixing. The damage suffered by coated particles was also investigated. The fractional failure of coatings during the mixing stage was negligible, about 0.5×10^{-4} , but during the subsequent hot pressing the factor rose to between 10×10^{-4} and 30×10^{-4} , according to the method of mixing. Failure of the coatings is probably due to friction at die walls during pressing or extraction of the compact. The provision of an unfueled zone surrounding the compacts reduced the damage factors to 5×10^{-4} , and further improvement is considered possible. (auth)

9.

13226 (ORNL-3970, pp 150-63) GAS-COOLED REACTOR PROGRAM. Coobs, J. H.; Harms, W. O. (Oak Ridge National Lab., Tenn.).

Techniques are being developed for fluidized-bed coating of particles and preparing and testing pyrolytic-C-coated particles and fueled-graphite elements containing such particles. The technology for deposition of coatings from CH_4 was extended to larger equipment and to the coating of larger ($\sim 500\text{-}\mu$ -dia) fuel particles. Techniques were also developed for depositing and characterizing low-density ($<1\text{ g/cm}^3$) coatings from C_2H_2 . Under heat treatment and irradiation, coated oxide fuels were stable and performed at least as well as if not better than coated carbides. Predictions based on a mathematical model for coated-particle behavior agreed well with actual results of a recent survey irradiation experiment. The model has also been valuable for planning further irradiation experiments and for optimizing the thickness ratios of two-layer coatings. Techniques were developed for depositing an optimum coating on oxide fuel particles and about 1.5 kg of coated $(\text{Th,U})\text{O}_2$ microspheres were prepared for a cooperative irradiation experiment in the Dragon Reactor in England. Heat-transfer analyses of the recent series of irradiation experiments on 6-cm-dia fueled-graphite spheres were also performed to correlate heat-generation rates with measured temperatures. Two processes for fabricating fission-product-emitting fueled-graphite elements are being evaluated as well. Other supporting activities included two experiments at higher temperatures on the creep behavior of graphite under irradiation. Results were consistent with a model for irradiation-induced creep and indicated that if the creep-rate coefficient is such that the stress level does not exceed the fracture stress, very large creep strains ($>2\%$) may be accommodated. The available thermodynamic data for ceramic nuclear fuels were analyzed and correlated. The metallurgical behavior of stainless steel and Inconel nozzles designed for temperature measurement in the EGCR was monitored in a series of mockup tests. In addition, the failure of a 42-in.-dia stainless steel bellows at the EGCR was analyzed. (auth)

10.

30870 (ORNL-TM-1528) COATED-PARTICLE FUELS DEVELOPMENT AT OAK RIDGE NATIONAL LABORATORY FOR PERIOD NOVEMBER 15, 1965 TO MAY 15, 1966. Coobs, J. H.; Morgan, J. G. (comps.) (Oak Ridge National Lab., Tenn.). May 1966. Contract W-7405-eng-26. 48p. Dep. mn. CFSTI \$2.00 cy. \$0.50 mn.

Highlights of the development work during the period Nov. 15, 1965, to May 15, 1966, on graphite-base fuel concepts involving pyrolytic-carbon-coated fuel particles for gas-cooled, civilian power reactor applications are reported. The major emphasis was on the testing and evaluation of pyrolytic-carbon-coated oxide fuel particles. A mathematical model for coated-particle behavior was used to guide the irradiation program and to correlate

results from irradiation experiments. The work described includes out-of-reactor characterization of coated-particle materials, irradiation testing, and postirradiation examination. (W.L.G.)

11.

13215 (ORNL-3885, pp 3-86) FUELED-GRAPHITE DEVELOPMENT. Coobs, J. H.; Sisman, O.; McCurdy, H. C. (Oak Ridge National Lab., Tenn.).

Methods for coating pyrolytic carbon to UC_2 and ThC_2 fuel particles are discussed in some detail. The results of a phase study of the pseudobinary system $\text{UC}_2\text{-ThC}_2$ are presented. Tabulations of thermodynamic data for the UC and UC_2 systems are given. Techniques for converting sol-gel oxide particles to carbides are discussed. Heat and irradiation tests were performed on the pyrolytic carbon coated fuel particle. Results from these tests are presented. Fuel assemblies employing carbon coated fuels were tested. (M.O.W.)

12.

13227 (ORNL-3970, pp 209-22) THORIUM UTILIZATION. Douglas, D. A. Jr. (Oak Ridge National Lab., Tenn.).

The activities reported involve characterization of the fuel, testing of the fuel under irradiation, and the development of processes for fabricating fuel elements prototypic of designs useful for power reactors. Since the ^{233}U bred from Th carries α and γ activity, personnel must be shielded during processing steps. An \$8,000,000 research facility in which pilot-scale processes can be demonstrated was designed and will be completed in December 1966. Remotely operated equipment is under development and will be installed and jointly operated. The equipment will be designed to prepare and encapsulate U and Th compounds in a graphite matrix for use in high-temperature gas-cooled reactors. Work was done to optimize the most critical step in the fabrication process, that of placing a carbon coating on microspheres of UO_2 and ThO_2 . The pyrolytically deposited C must meet several density and anisotropy requirements in addition to specifications for sphericity and thickness. The significant process variables in a fluidized-bed coater have been identified and are being quantified so that a completely automatic system can be designed. As an alternate approach to the problem, a heated rotating drum could be employed as a coating system. Much larger charges of particles are possible in such a device, but the low efficiency of the gas-solid contact is a major obstacle that has not yet been solved. Although the concept of using carbon-coated oxides has proven very satisfactory, the desire to use part of the fuel in the carbide form persists. Although the carbide can be made by mixing of solids, a higher density product can be achieved by the process of solid-solid conversion demonstrated last year. A pilot-scale version of the process has been designed and fabricated. The irradiation program has two objectives. One is to explore the capabilities of ThO_2 -base fuels so that their full potentialities can be utilized by designers, and the second is to proof test the products from the processes and equipment designed for remote operation. Most of the accomplishments are related to the first objective. ThO_2 -base fuels are stable in performance at exposures of over 1000 reactor full-power days. Successful tests were completed with heat ratings of 1000 W/cm. It is concluded that ThO_2 -base fuels have marked superiority to UO_2 fuels in reactivity control and structural stability at very high heat ratings. Irradiation of coated microspheres of sol-gel-derived fuel is a joint undertaking with the ORNL Gas-Cooled Reactor Program. Results indicate that sol-gel-prepared fuels perform equivalently to ceramics processed by older methods. Carbon-coated oxide fuels have performed satisfactorily despite thermodynamic instability. Sol-gel techniques provide extra degrees of flexibility in the forming of ceramic bodies. In the gel state, extrusion without the use of plasticizers has been demonstrated. Packing of preselected sized microspheres to 85% of theoretical density in small-diameter

tubes has been accomplished with very low-energy vibration. A basic study of the sintering and grain growth of ThO_2 and UO_2 produced by the sol-gel process uses differential thermal analyses and hot stage microscopy. Although no simple law seems to describe the rate effects observed, shrinkage and crystal growth are evidently related to crystallite size. (auth)

13.

35867 (TRG-Report-1180) PYROLYTIC-CARBON-COATED URANIUM CARBIDE PARTICLES. THE EFFECT OF DEPOSITION CONDITIONS ON COATING EFFICIENCY, STRUCTURE, AND DENSITY. Ford, L. H.; Ingleby, B. E.; Hibbert, N. S. (Reactor Group, United Kingdom Atomic Energy Authority, Springfield (England)). Jan. 24, 1966. 25p. Dep. BIS \$1.00. HMSO 5s.

The efficiency of pyrolytic C coating deposition on UC particles was shown to vary directly with the surface area of the bed and temperature of deposition, and inversely with increase in CH_4 partial pressure and degree of fluidization. Variations of coating structure were classified into five groups, four according to the degree of isotropy and the size of the grain structure observed, and the fifth by the absence of any visible structure. For the equipment used, the structure of the coatings was found to vary with the surface area of the charge, the temperature of coating, gas composition, and the degree of fluidization. Measurements of coating density show the well known minimum density region in the temperature range 1400 to 1600°C but independently of coating structure. (auth)

14.

42624 (GA-6975) THE HTGR COATED PARTICLE FUEL. Goeddel, W. V.; Bokros, J. C. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Mar. 15, 1966. Contract AT(04-3)-167. 28p. (CONF-661003-3). Dep. mn. CFSTI \$2.00 cy, \$0.50 mn.

For Presentation at AIME Nuclear Metallurgy Symposium on High Temperature Nuclear Fuels, Delevan, Wis.

Comprehensive studies of coated particle fuels for the HTGR (High-Temperature Gas-cooled Reactor) are described. These studies are directed toward identifying and quantitatively defining the structural and environmental parameters influencing the behavior of pyrolytic-carbon-coated carbide fuel particles under conditions of high temperature, high burnup, and fast flux irradiation. Systematic studies of the relationship between the structure of pyrolytic carbon and its behavior under irradiation show that the most irradiation-stable carbon coatings will be those that are free of any preferred orientation, have densities as near theoretical as possible, and have maximum crystallinity to minimize irradiation-induced crystallite strains. Extensive irradiation testing of pyrolytic-carbon-coated carbide fuel particles shows that their stability is dependent on both environmental conditions (temperature and burnup) and material parameters (coating structure and geometry). Two- and three-layer pyrolytic-carbon coatings that have a low-density innermost layer and high-density isotropic outer layers have shown good stability during 10% to over 20% burnup at temperatures to over 1400°C. (auth)

15.

35834 (BMI-1767) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING MARCH, 1966. Keller, Donald L. (Battelle Memorial Inst., Columbus, Ohio). Apr. 1, 1966. Contract W-7405-eng-92. 37p. Dep. mn. CFSTI \$2.00 cy, \$0.50 mn.

Research progress on reactor materials and components, fuels, UN-PuN fuel materials, a portable smoke photometer, coated-particle fuel materials, and corrosion studies of fluidized-bed fluoride volatility processes. (M.C.G.)

16.

37168 (BMI-1771) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING APRIL 1966. Keller, Donald L. (Battelle Memorial Inst., Columbus, Ohio). May 1, 1966. Contract W-7405-eng-92. 33p. Dep. mn. CFSTI \$2.00 cy, \$0.50 mn.

Research progress is reported in sections on reactor materials and components, ceramic and cermet fuels, UN-PUN fuel materials, portable smoke photometer, coated particle fuel materials, and corrosion resistance of construction materials used in fluidized-bed fluoride volatility processes for fuel recovery. (J.R.D.)

17.

18308 (BMI-1784) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING JULY THROUGH SEPTEMBER 1966. Keller, Donald L. (Battelle Memorial Inst., Columbus, Ohio). Oct. 1, 1966. Contract W-7405-eng-92. 56p. Dep. mn. CFSTI \$3.00 cy, \$0.65 mn.

The following topics are covered: UN-PuN fuel materials, irradiation effects on reactor cladding materials, coated-particle fuel materials (ceramic-coated Pu-base), pyrolytic C-coated UN, and graphite coatings, development of $\text{BeO-ThO}_2\text{-PuO}_2$ fuels for gas-cooled reactors, $\text{UO}_2\text{-PuO}_2$ fuel development, and effects of high burnup on $\text{UO}_2\text{-CeO}_2$ and $\text{UO}_2\text{-ZrO}_2$ fuels. (D.L.C.)

18.

43860 (SGAE-CH-35/1966) FABRICATION METHODS AND EVALUATION OF URANIUM-THORIUM-CARBIDE FUEL FOR HIGH TEMPERATURE GAS COOLED REACTORS. Koss, P.; Bildstein, H. (Oesterreichische Studiengesellschaft fuer Atomenergie G.m.b.H., Seibersdorf). [1966]. 31p. (SGAE-M-5/1966; CONF-660524-12). Dep.

From 2nd International Thorium Fuel Cycle Symposium, Gatlinburg, Tenn.

A discussion is presented of the development work carried out on the preparation and reprocessing of coated fuel particles for HTGR. The paraffin and resin processes used in the production of kernels are described. Equipment and processes for coating the particles with pyrocarbon are discussed. The testing and evaluation methods include surface contamination determination and broken particle detection by chlorination. (D.L.C.)

19.

5014 (GA-6356) DESCRIPTION AND RESULTS OF THE TARGET P7 COATED PARTICLE IRRADIATION CAPSULE. Luby, C. S. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). June 22, 1966. Contract AT(04-3)-167. 36p. Dep. mn. CFSTI \$3.00 cy, \$0.65 mn. Supersedes GAMD-6366, dated May 17, 1965.

Twelve types of coated fuel particles were irradiated to 16 to 24% burnup at 1000 to 1250°C. Post-irradiation examination indicated that coating failure increased with decreasing coating thickness and/or increasing particle size. A coating-thickness to particle-dia ratio of 0.4 or higher appeared to be necessary for radiation stability. Buffer-isotropic coatings appeared equivalent in radiation stability to Triplex coatings, and coated oxides appeared equivalent to coated carbides. (auth)

20.

1957 (GA-6493) DESCRIPTION AND RESULTS OF THE TARGET P8 COATED PARTICLE IRRADIATION CAPSULE. Luby, C. S. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). June 22, 1966. Contract AT(04-3)-167. 44p. Dep. mn. CFSTI \$2.00 cy, \$0.50 mn.

Twelve types of coated fuel particles were irradiated to 11 to 17% burnup at 900 to 1500°C. Post-irradiation examination indicated that Buffer-isotropic coatings appear better than Triplex coatings in radiation stability, and coated oxides appear slightly better than coated carbides. Pre-irradiation thermal treatment had no effect on the radiation stability of coated oxide particles, but a slight detrimental effect on coated carbides. The relatively high failure rate of many particles in this test was attributed in major part to a low coating-thickness to fuel-particle-dia ratio (0.25). (auth)

21.

22935 (GA-7171) DESCRIPTION AND RESULTS OF THE P10 COATED PARTICLE IRRADIATION EXPERIMENT. Luby, C. S. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). July 21, 1966. Contract AT(04-3)-167. 50p. Dep. mn. CFSTI \$3.00 cy, \$0.65 mn.

Twelve types of coated fuel particles were irradiated to 9 to 16% burnup (FIMA) at temperatures of 950 to 1700°C (average = 1330°C) in Capsule P10. The fast flux exposure was 7.0×10^{20} nvt ($E > 0.18$ MeV). Postirradiation examination indicated that the stability of isotropic pyrolytic carbon coatings increased with increasing coating density and that small fuel particles had greater stability than large particles (with constant coating-thickness to fuel-particle-diameter ratio). (auth)

22.

6892 (GA-7286) DESCRIPTION AND RESULTS OF THE P11 COATED PARTICLE IRRADIATION EXPERIMENT. Luby, C. S.; Chin, J.; Spillane, B. B. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Oct. 27, 1966. Contract AT(04-3)-167. 47p. Dep. mn. CFSTI \$3.00 cy, \$0.65 mn.

In the P11 capsule experiment, several types of improved coated fuel particles were irradiated at high temperatures (1170 to 1560°C) to 1 to 20% burnup (FIMA). Coated UC₂ particles and (Th,U)C₂ particles with different Th:U ratios (0:1, 0.4:1, 1:1, and 3:1) were used to achieve burnup rates of 15, 10, 7, and 4% per month, respectively. The fast-neutron-flux exposure was 4.7×10^{20} nvt ($E > 0.18$ MeV). The results of postirradiation examination indicate that (1) the Buffer-isotropic (BISO) coated and the Triplex coated particles survive very high burnups (at highly accelerated burnup rates) at and above the anticipated temperatures of the HTGR, and that (2) burnup rate and irradiation temperature significantly affect the irradiation stability of coated particles. (auth)

23.

8147 INFLUENCE OF IRRADIATION TEMPERATURE ON THE IRRADIATION STABILITY OF COATED-PARTICLE FUEL. Luby, C. S.; Gerdlel, W. V. (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Trans. Amer. Nucl. Soc., 9: 422(Oct.-Nov. 1966).

24.

6883 (INR-753/IV,II/C) THE STRUCTURE OF PYROLYTIC CARBON COATINGS ON URANIUM DICARBIDE PARTICLES. Mucha, Franciszek; Szmid, Zofia; Kallnowski, Bohdan (Institute of Nuclear Research, Warsaw (Poland)). Aug. 1966. 20p. Dep.

X-ray studies on the structure of pyrolytic carbons, deposited on UC₂ particles by thermal decomposition of CH₄ in a fluidized-bed, have been carried out. The influence of the temperature of

pyrolysis at 1400 to 2150°C, and the concentration of CH₄ in the range 0.05 to 1.00 l CH₄/l [CH₄,Ar] has been examined. The structure of pyrolytic carbons depends largely on these parameters. The degree of ordering of the C structure increases with the increase in the temperature, and with the decrease in the concentration of CH₄. (auth)

25.

40385 (ORNL-P-2327) CURRENT STATUS OF IRRADIATION TESTING OF THORIUM FUELS AT OAK RIDGE NATIONAL LABORATORY. Olsen, A. R.; Coobs, J. H.; Ullmann, J. W. (Oak Ridge National Lab., Tenn.). [1966]. Contract W-7405-eng-26. 30p. (CONF-660524-10). Dep. mn. CFSTI \$2.00 cy, \$0.50 mn.

From 2nd International Thorium Fuel Cycle Symposium, Gatlinburg, Tenn.

The sol-gel process production, fabrication procedures, and irradiation tests are summarized for pure or mixed compounds of Th, U, and Pu as oxides and carbides. Materials characteristics and irradiation behavior are described for both vibrantly compacted metal-clad bulk oxide fuels and pyrolytic carbon coated particle fuels. (H.D.R.)

26.

18076 (ORNL-TM-1385) A COMPUTER PROGRAM FOR PREDICTING COATED-PARTICLE BEHAVIOR. Prados, J. W. (Oak Ridge National Lab., Tenn.). Mar. 1966. Contract W-7405-eng-26. 32p. Dep. mn. CFSTI \$2.00 cy, \$0.50 mn.

A computer program is developed for use in calculating conditions under which the pyrolytic-carbon coatings of fuel microspheres would be expected to fail. Working equations and a listing of the Fortran-63 source program are presented, together with instructions for furnishing input data. (auth)

27.

20220 (ORNL-TM-1358) NUFTIC—AN INFORMATION CENTER ON NUCLEAR FUEL TECHNOLOGY. Pilloton, R. L. (Oak Ridge National Lab., Tenn.). Mar. 1966. Contract W-7405-eng-26. 35p. Dep. mn. CFSTI \$2.00 cy, \$0.50 mn.

The amount of literature on the fabrication, evaluation, application, and economics of nuclear fuels is increasing rapidly. To cope with the difficulty of keeping informed, the Metals and Ceramics Division of the Oak Ridge National Laboratory has organized the Nuclear Fuel Technology Information Center (NUFTIC). In this center active scientists and engineers review the literature. A computer stores, searches, and prints information. The choice of information processing system, the method of operation of the center, and examples of the output are described. (auth)

28.

31795 STUDY OF IRRADIATED AND UNIRRADIATED COATED FUEL PARTICLES BY THE ELECTRON MICRO-PROBE. Quatart, D.; Schleicher, H. W. (EURATOM CCR, Ispra, Italy). J. Nucl. Mater., 19: 221-33(June 1966).

Methods of examination of coated particle nuclear fuel by the electron microprobe are described. The determination of low concentrations of uranium in graphite and the analy-

sis of highly radioactive, fully irradiated particles necessitated special alterations to equipment and procedures which resulted in a remarkable improvement in precision. Examples of the results obtained on different types of particles are presented. (auth)

29.

33957 PERFORMANCE OF PYROLYTIC-CARBON-COATED URANIUM-OXIDE PARTICLES DURING IRRADIATION AT HIGH TEMPERATURES. Reagan, P. E.; Morgan, J. G.; Sisman, O. (Oak Ridge National Lab., Tenn.). Trans. Amer. Nucl. Soc., 9: 28-9 (June 1966).

30.

13261 (ORNL-3885, pp 87-149) INVESTIGATIONS OF FUELED-GRAPHITE SYSTEMS. Sisman, O.; Savage, H. W.; Watson, G. M. (Oak Ridge National Lab., Tenn.). Discussions are presented on instantaneous fission gas release experiments, fission product release from high-burnup coated UC_2 , $(Th,U)C_2$, and UO_2 particles by post-irradiation annealing, sorption of fission product vapors on structural metals, compatibility of pyrolytic carbon-coated fuel particles with water vapor, and recoil of fission products in pyrolytic carbon. (M.O.W.)

31.

8148 THERMAL CONDUCTIVITY OF BEDS OF COATED FUEL PARTICLES. Stevens, D. W. (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Trans. Amer. Nucl. Soc., 9: 424-5 (Oct.-Nov. 1966).

32.

5089 (GA-6181) ADVANCED, GRAPHITE-MATRIX, DISPERSION-TYPE FUEL SYSTEMS AND RESEARCH ON GRAPHITE. Quarterly Progress Report for the Period Ending February 14, 1965. Bokros, J. C.; Goeddel, W. V.; Lonsdale, H. K.; Price, R. J.; White, J. L.; Zumwalt, L. R. (General Atomic, San Diego, Calif. John Jay Hopkins Lab. for Pure and Applied Science). Feb. 15, 1965. Contract AT(04-3)-167. 64p. Dep. mn; CFSTI \$3.00 cy, \$0.75 mn.

Pyrolytic-carbon coating studies were extended from 1900 to 2400°C. The densities of the structures deposited increased uniformly with deposition temperature. Both isotropic and granular structures were deposited; low methane concentrations, high temperatures, and small surface areas favored deposition of granular structures. Mechanical property measurements performed on the high-temperature deposits showed that the modulus, fracture stress, and fracture strain increase with increasing deposition temperature for granular structures. Isotropic and transition structures showed a slight decrease in modulus and fracture stress with increasing deposition temperature. Newly developed x-ray scattering techniques were applied to carbons deposited at 1400°C from 5 and 40% methane. Measurement of the physical and mechanical properties of annealed laminar structures, which were deposited at 1400°C under various conditions, showed that while various properties respond differently to annealing, the order of annealability between structures remains the same. In general, the annealability increases with a decrease in the amount of single layers in a structure. Tracer-level diffusion studies conducted on granular pyrolytic carbon structures using ^{228}Th as the diffusive species showed that the diffusion coefficients for this type structure are about

a factor of 10 greater than for laminar pyrolytic carbon. These results confirmed that the diffusion coefficient is strongly dependent on the structure of the pyrolytic carbon. Studies of the influence of additives dispersed within a graphite matrix on the hot-working process and on the properties of the hot-worked bodies resulted in the development of unique carbide-graphite cermets. These new materials were prepared by hot-working carbide-bearing graphite bodies at temperatures such that the dispersed carbide phase is in the liquid state. Hot-working of graphite bodies containing MoC or ZrC in the liquid state resulted in large increases in compressive strengths parallel to the direction of hot-working, and modest increases in the perpendicular direction. The helium permeability of these carbide-graphite cermets is near the limit of detection with existing apparatus ($<10^{-4}$ cm²/sec). (P.C.H.)

33.

40592 IMPROVEMENTS IN OR RELATING TO COATED ARTICLES. (to General Dynamics Corp.). British Patent 998,247. July 14, 1965. Priority date Apr. 11, 1963, United States.

A coated article is provided that has increased structural stability under conditions of thermal and/or radiation stress. The article includes a core, an exterior coating about the core, and a low density, spongy, shock absorbing coating capable of absorbing stresses interposed between the core and the exterior coating. The coated article may be a particular fissionable material or a poison, and the shock absorbing coating may be a low density, spongy, pyrolytic carbon. (B.L.M.)

34.

29616 (BMI-1738) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING JULY, 1965. Dayton, Russell W.; Paprocki, Stan J. (Battelle Memorial Inst., Columbus, Ohio). Aug. 1, 1965. Contract W-7405-eng-92. 39p. Dep. mn. CFSTI \$1.00 cy, \$0.50 mn.

Research and development progress is reported on reactor materials and components, ceramic fuels, radioisotope and radiation applications, coated particle fuels, fluoride volatility processes, and gas cooled reactor materials. (J.R.D.)

35.

16127 RESEARCH CENTERS IN EUROPE. NO. 12. SEIBERSDORF. Euronuclear, 2: 607-14 (Dec. 1965).

A description of the organization and work at the Seibersdorf center is given. Most of the work is of a fundamental nature. But work was also done on coated fuel particles for the DRAGON project of the ENEA. The work of the IAEA laboratory consists for the most part in providing highly specialized services for member countries in the fields of radioactive analysis and standardization, as well as studies, and co-ordination work on agricultural and medical applications of nuclear radiations. Future projects at Seibersdorf are also mentioned. (J.F.P.)

36.

11343 (LA-3367) THE BEHAVIOR OF COATED PARTICLES IN A LARGE NUCLEAR TRANSIENT (KIWI TNT). Geer, W. U.; Hoffman, C. G.; Kerrisk, J. F. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). Sept. 1, 1965. Contract W-7405-eng-36. 78p. Dep. mn. CFSTI \$3.00 cy, \$0.75 mn.

Capsules containing extruded graphite fuel samples, loaded with 19 different types of coated particles, were exposed to the neutron pulse generated by the Kiwi Transient Nuclear Test (Kiwi TNT). The effects of flux, particle core material (UC_2 and UO_2), core diameter, coating thickness, and sample temperature on the graphite matrix damage and the coated particle damage were investigated. Visual, metallographic, and radiographic examination of the recovered samples produced the following conclusions: the total energy deposition was by far the most significant variable giving complete destruction of the samples at the highest energy depositions and little or no change at the lowest; particles with large core sizes showed more damage than those with small cores; neither coating thickness nor sample temperature significantly affected damage; and particles with UO_2 cores were more easily damaged than those with UC_2 cores. (auth)

37.

23742 (GA-6418) TARGET—A PROGRAM FOR A 1000-Mw(e) HIGH-TEMPERATURE GAS-COOLED REACTOR. Quarterly Progress Report for the Period Ending May 31, 1965. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). June 30, 1965. Contract AT(04-3)-167. 178p. Dep. mn. CFSTI \$5.00 cy, \$1.00 mn.

Efforts devoted to fuel development are reported in the areas of mechanical design, materials technology, fission product behavior, GAIL loop operation, recycle technology, and reactor core analyses. Development and analysis of prestressed concrete reactor vessels are summarized. (J.R.D.)

38.

23743 (GA-6671) HTGR BASE PROGRAM. Quarterly Progress Report for the Period Ending August 31, 1965. (General Dynamics Corp., San Diego, Calif. General Atomic Div.). Sept. 30, 1965. Contract AT(04-3)-167. 111p. Dep. mn. CFSTI \$4.00 cy, \$0.75 mn.

Activities in a program directed toward design and development of a 1000-Mw(e) gas-cooled reactor are reported. Progress is summarized on development of prestressed concrete reactor vessels, in-pile loop testing, reactor physics methods development, fuel recycle studies, and design of advanced fuels. (J.R.D.)

39.

39129 (ORNL-TM-1116) COATED-PARTICLE FUELS DEVELOPMENT AT OAK RIDGE NATIONAL LABORATORY FOR PERIOD NOVEMBER 15, 1964 TO MAY 15, 1965. Harms, W. O.; Coobs, J. H. (comps.) (Oak Ridge National Lab., Tenn.). May 1965. Contract W-7405-eng-26. 102p. Dep. mn; CFSTI, \$4.00 cy, \$0.75 mn.

Development work on graphite-base fuels for gas-cooled reactors is summarized. Information is included on pyrolytic carbon coatings, testing of coated particles, properties and testing of pyrolytic carbon, coated graphite fuel particles, and irradiation testing. Research on processing of high-temperature gas-cooled reactor fuels is described. (J.R.D.)

40.

25707 (ORNL-TM-1123) FABRICATION VARIABLES, PERFORMANCE, AND COST CONSIDERATIONS FOR HTGR COATED-PARTICLE FUELS. W. O. Harms and D. B.

Trauger (Oak Ridge National Lab., Tenn.). Apr. 30, 1965. Contract W-7405-eng-26. 31p. (CONF-650606-5). Dep. (mn); \$2.00(cy), 1(mn) CFSTI.

From EURATOM Symposium on Fuel Cycles of High-Temperature Gas-Cooled Reactors, Brussels.

Studies involving the development of pyrolytic-carbon-coated uranium-thorium carbide particles and of graphite-matrix fuels containing coated particles have produced a basis for HTGR fuel element designs and for fuel-cycle cost analysis. Irradiation tests combined with thermal and mechanical test evaluations have demonstrated satisfactory fuel element performance for pebble-bed reactor application. More recent work on particle coating and evaluation techniques has led to means for improved control of production variables. Pyrolytic-carbon-coated oxide fuel particles offer promise for reduced costs and satisfactory reactor service. The results presented are generally applicable to prismatic fuel structures as well as to spheres. (auth)

41.

26888 IRRADIATION STUDIES OF LEAKPROOF (COATED PARTICLE) FUELS FOR THE DRAGON REACTOR. Gilles Hauser. Energie Nucl., 7: 100-12 (Mar.-Apr. 1965). (In French)

Irradiation experiments were made on coated-particle fuels. A great number of the particle parameters were varied for the purpose of this study. Kernels contained $(Th,U)C_2$ or $(Zr,U)C$ solid solutions of variable compositions. Single or multi-layer pyrocarbon coatings were used, with or without a layer of silicon carbide. Irradiation conditions (temperature, burnup) were also varied in several facilities either sealed or purged. Results showed that it is possible to get very low fission gas releases ($R/B = 10^{-4}$) at temperatures up to 1600°C. As far as solid fission products are concerned, the best results were observed when using an intermediate layer of silicon carbide within the coating. No influence of the burnup was observed on the fuel behavior. (auth)

42.

556 (CONF-650407, pp 56-72) FISSION PRODUCT RELEASE FROM COATED PARTICLE FUELS UNDER ACCIDENT CONDITIONS. Morrison, D. L.; Rosenberg, H. S.; Townley, C. W.; Sunderman, D. N. (Battelle Memorial Inst., Columbus, Ohio).

During the development of the coated particle fuels, data on fission product release have been obtained in postirradiation experiments in helium atmospheres at temperatures up to 2000°C for low-burnup specimens. The data from these laboratory studies were examined to determine what effects pyrolytic carbon coatings surrounding the fuel would have upon the release of fission products from the fuel under accident conditions. Although the mechanisms governing the release in postirradiation heat treatments have not been established, some measurements of the time dependence and the temperature dependence have been reported. These data were used to estimate the extent of the release from the fuel following the loss of coolant. If the temperature distribution in the Peach Bottom reactor following a loss-of-coolant accident is assumed to be typical for gas-cooled reactors using coated particle fuels, an average temperature within the core of 1540°C would be expected for a period of 20 to 100 hr after the accident. During this period at the average temperature, a fractional release of krypton and xenon of 4×10^{-7} would be estimated for release from unsupported pyrolytic carbon-coated particles.

The fractional release of iodine is estimated at 3×10^{-4} , and the fractional release of barium and strontium would be 4×10^{-2} and 5×10^{-2} respectively. The elemental distribution of fission products released from the coated fuel particles under these conditions is substantially different from the expected distribution during the meltdown of a metal-clad, UO_2 -fueled core. The estimated fractional release of barium and strontium from coated particles is higher than that normally assumed under the guidelines of TID-14844 for release of solid fission products to the reactor building. The fractional releases estimated above were for low-burnup samples of unsupported coated particles. The effects of burnup on the fractional release must be evaluated, and since most fuel concepts employ the particles dispersed in a graphite matrix or surrounded by graphite, any retention of the fission products by the graphite bodies would reduce the expected release. Although the barium and strontium release from the coated fuel particles is high, the loss of these metals or their oxides from the core during the accident may be controlled by evaporation or other processes. The possible alterations to the estimated release by these factors based upon the available data in these areas are discussed. (D.L.C.)

43.

24922 (ORNL-3731(p.1-110)) ADVANCED GAS-COOLED REACTOR PROGRAM. (Oak Ridge National Lab., Tenn.).

Studies of the effects of process conditions on the properties of pyrolytic C fuel coatings were continued. In experiments in which deposition temperatures and methane pressures were varied, the microhardness of coatings increased with increasing methane pressures. The physical structure of coatings was studied by x-ray diffraction and electron microscopy. Results of coated-fuel-particle density and shape investigations are included. Phase studies of Th-U-C systems are discussed. Results of fueled graphite sphere impact testing are discussed along with results of irradiation tests. (J.R.D.)

44.

7500 (ORNL-TM-1335) MACHINE-ASSEMBLED REVIEWS ON COATED NUCLEAR FUEL PARTICLES. Pilloton, R. L. (Oak Ridge National Lab., Tenn.). Dec. 1965. Contract W-7405-eng-26. 80p. Dep. mn; CFSTI \$3.00 cy, \$0.75 mn.

A computer-produced selection of approximately 425 reviews of recent reports, articles, and patents on coated nuclear fuel particles is presented. The 50- to 150-word summaries were prepared by the Nuclear Fuel Technology Information Center and extracted from a general magnetic tape file by a computer program based on reference codes and keywords. Specific information on the preparation, evaluation, applications, and economics of coated-particle fuels is given along with general information on processes widely used in the field such as fluidization. Subject and author indexes are included. (P.C.H.)

45.

34653 (GA-6399) FUEL FABRICATION TECHNIQUES FOR HIGH TEMPERATURE GAS-COOLED REACTOR FUELS. Ragone, D. V. (General Atomic, San Diego, Calif. John Jay Hopkins Lab. for Pure and Applied Science). May 7, 1965. Contract AT(04-3)-167. 18p. (CONF-650606-8) Dep.(mn); \$1.00(cy), 1(mn) CFSTI.

From EURATOM Symposium on Fuel Cycles of High-

Temperature Gas-Cooled Reactors, Brussels.

The fuel for future General Atomic High Temperature Gas-Cooled Reactors will probably be in the form of fission-product-retaining, coated particles. These will be incorporated, either loose or bonded, into a graphite structure which will serve as the fuel element. The fabrication of these coated particles consists of two steps, the formation of the fuel-bearing seed and the deposition of the coating. Relative merits of "dry" processes and "wet" processes for seed production are discussed. "Dry" processes include agglomeration, followed by sintering or fusion of the agglomerate, and plasma arc processes. The sol gel process is used as an example of a "wet" technique. Of the techniques listed, the agglomeration-sinter or melt combination is the method by which most of the particles to date have been made, while the sol gel method holds the most promise for low cost fabrication. The seeds are coated by the application of pyrocarbon and/or silicon carbide by pyrolysis of appropriate compounds while the particles are suspended or agitated in some way. This can be done in a fluidized bed, a rotating drum device (similar to a ball mill) or in a vibrating bed. The fluidized bed technique is probably the most developed of the ones listed. Coatings having a great variety of physical and mechanical properties can be deposited by this method by varying the temperature of deposition, the gas flow rate and the coating gas composition. There are still a great number of uncertainties in the production processes which can only be resolved by production experience. (auth)

46.

899 FISSION-GAS RELEASE FROM PYROLYTIC-CARBON-COATED FUEL PARTICLES DURING IRRADIATION AT 2000 TO 2500°F. Reagan, P. E.; Morgan, J. G.; Sisman, O. (Oak Ridge National Lab., Tenn.). Nucl. Sci. Eng., 23: 215-33(Nov. 1965).

The fission-gas release from pyrolytic-carbon-coated fuel particles during irradiation was studied for gas-cooled reactor application. Duplex- and triplex-type coatings on thorium-uranium carbide cores and on uranium carbide cores were tested at temperatures between 2000 and 2500°F (1093 and 1371°C). Both types of coatings retained fission gas quite well up to about 20 at.% heavy-metal burnup. Postirradiation examination revealed that the particles with the duplex coating were more susceptible to radiation damage (by the formation of a reaction zone at the core/coating interface) than were the particles with the triplex coating. This damage, however, did not affect the fission-gas release rates. (auth)

47.

44740 (BNI-1737) IN-PILE STUDIES OF EXPERIMENTAL PYROLYTIC-CARBON-COATED UC_2 AND UO_2 FUEL PARTICLES. Ritzman, Robert L.; Martin, Robert L.; Miller, Neil E.; Townley, Charles W. (Battelle Memorial Inst., Columbus, Ohio). July 28, 1965. Contract W-7405-eng-92. 52p. Dep. mn; CFSTI \$1.25 cy, \$0.50 mn.

The irradiation performance of pyrolytic-carbon-coated UC_2 and UO_2 particles was studied at temperatures ranging from 100 to 1300°C and to fuel burnups of 12 at. % of the ^{235}U . It was found that coatings on high-density UC_2 particles must be greater than 110 μ thick in order to survive low-temperature irradiation, and in order to exhibit high integrity at elevated temperatures, the coatings must be multilayered as well. However, by using porous UO_2 fuel particles (~80 per cent dense), pyrolytic-carbon coatings

as thin as 50 μ performed without failure at low temperature, and a 73 μ columnar coating exhibited only slight failure when irradiated to a ^{235}U burnup of 8.4 at. % at 1300°C. Large (460 μ in diameter) porous UO_2 particles with 113- μ laminar pyrolytic-carbon coatings, currently under study sustained burnups of over 7.5 at. % at 815°C with no indication of coating fracture. Failure at both low and high temperatures is believed to be due primarily to changes induced by fission-fragment recoil in both the fuel-core and coating materials, resulting in stresses in the pyrolytic carbon sufficient to crack the coatings. The in-pile studies were performed in both static and sweep-gas capsules with the particles held in a graphite matrix or in the unsupported condition. Postirradiation examinations included fission-gas-release measurements, determination of the fraction of exposed uranium, and metallographic examination. (auth)

48.

39228 (BMI-1734) POSTIRRADIATION FISSION-PRODUCT RELEASE FROM COATED FUEL PARTICLES. Rosenberg, Harvey S.; Morrison, David L.; Townley, Charles W.; Sunderman, Duane N. (Battelle Memorial Inst., Columbus, Ohio). July 7, 1965. Contract W-7405-eng-92. 52p. Dep. mn; CFSTI, \$1.25 cy, \$0.50 mn.

For PyC-coated particles, the release of ^{140}Ba , ^{89}Sr , and ^{111}Ag was greater than 1000 ppm, the release of ^{132}Te , ^{131}I , ^{141}Ce , and ^{147}Nd was between 10 and 1000 ppm, and the release of ^{133}Xe , ^{103}Ru , and ^{95}Zr was less than 10 ppm for 6 hr of annealing at 1150°C. At a constant coating density the ^{140}Ba release was lowered by about two orders of magnitude as the thickness of the PyC coating was increased from 30 to 75 μ , and at a constant coating thickness the ^{140}Ba release was lowered by about one order of magnitude as the coating density was increased from 1.5 to 2.0 g per cm^3 . The ^{140}Ba release was lowered by three orders of magnitude when a SiC barrier layer was present in the PyC coating. The time dependence of release is very complex and varies from one fission product to another. The release rate is increased by a factor of about six for each 150°C increase in temperature between 1000 and 1300°C. The mechanism of release is thought to involve diffusion and compound formation between the fission-product elements and the particle coating. The apparent diffusion coefficients in one type of PyC at 1150°C are $9 \times 10^{-9} \text{ cm}^2$ per sec for ^{89}Sr , $6 \times 10^{-10} \text{ cm}^2$ per sec for ^{140}Ba , $5 \times 10^{-11} \text{ cm}^2$ per sec for ^{147}Nd , and $1.5 \times 10^{-11} \text{ cm}^2$ per sec for ^{141}Ce . (auth)

49.

36960 (ORNL-3807, pp 80-142) INVESTIGATIONS OF FUELED-GRAPHITE SYSTEMS. Sisman, O.; Savage, H. W.; Watson, G. M. (Oak Ridge National Lab., Tenn.).

From steady-state experiments with single-crystal UO_2 , a defect-trap model for fission-gas release was evolved. A basic assumption for this model is that defects such as grain boundaries will trap fission gases. A mathematical expression for the defect-trap model was derived and in order to evaluate the parameters in the equation, an experiment to allow slight oscillations of temperature and fission rate is in progress. The time-dependent responses of the fission gas to these oscillations are being used to evaluate the mathematical expression. The fractional fission-gas release and postirradiation examination results for nine experimental assemblies containing pyrolytic-carbon-coated fuel particles irradiated to a burnup of 7 at. % heavy metal at 1400°C were obtained. The coatings were on thorium-uranium carbide or uranium oxide cores. Damage to pyrolytic C coatings occurred because of melting

of a thermocouple sheath alloy and because the sweep gas became contaminated. Postirradiation anneals on nine groups of pyrolytic-carbon-coated fuel particles were completed. Samples irradiated up to 26% uranium-metal burnup were heated to temperatures up to 2000°C, usually for periods of 7 hr. Coating failures were generally less than 2%. The release of ^{85}Kr was a few ppm/hr through unbroken coatings. Solid fission-product release, especially Ba and Sr, was much higher; up to 0.6% per hr at 1370°C, up to 3% per hr at 1700°C, and more than 2% per hr at 2000°C. Ce and Cs release rates were significant, and Ru and Zr were also found. The data from fission product transport studies indicate that I deposition is a function of the material of the deposition surface. Investigations of problems related to contamination of metal surfaces by gas-borne fission products were continued. In studies of actinide diffusion in pyrolytic C, the thin-layer and constant-potential techniques were used to investigate the effects of initial actinide concentration and pyrolytic-C structure and the phenomena of segregation of actinides during diffusion. Autoradiography of specimens from the thin-layer experiments provided an explanation for the nonuniform actinide penetration observed previously. A study of the recoil of fission products in pyrolytic C was initiated to determine fission-fragment distributions in pyrolytic-C targets after exposure to fragment beams. Preliminary data were compiled for the compatibility of various batches of pyrolytic-carbon-coated fuel particles with partial pressures of water vapor of approximately 45 and 565 torr at 1000°C for exposure periods of 1 to 4.5 hr. Reaction rates obtained from weight losses and the incidence of coating failures as determined from acid leaching of U and Th are included. (J.R.D.)

50.

15002 (LA-3378) COATED PARTICLE FUEL ELEMENTS FOR UHTREX. Taub, J. M.; Bard, R. J. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). July 1965. Contract W-7405-eng-36. 49p. Dep. mn. CFSTI \$2.00 cy, \$0.50 mn.

Fuel elements for the start-up and early operation of UHTREX contain coated UC_2 particles in a graphite matrix. The fuel particles have triplex (three-layer) pyrocarbon coatings. The fuel element shock, 1 in. OD $\times$ 0.5 in. ID, is fabricated by extruding a mixture of fuel particles, graphite flour, carbon black (Thermax), and resin binder (Varcum). The extruded stock is successively cured in air to 250°C, baked in vacuum to 800°C, heated in helium to 1700°C, and cut into 5.5-in.-long fuel elements. The fuel elements are then leached with 20% HCl in helium at 1600°C to remove uranium exposed by the slight amount of damage to fuel particles that occurs during the mixing and extrusion steps and by the cutting operation. The coated fuel particles, other fuel element ingredients, fabrication equipment and procedures, and properties of the finished fuel elements are described. (auth)

51.

39126 (CONF-117(p.149-70)) PREPARATION AND EVALUATION OF CARBON-COATED FUEL PARTICLES. Browning, M. F.; Secrest, A. C.; Secrest, V. M.; Landrigan, R. B.; Blocher, J. M. Jr. (Battelle Memorial Inst., Columbus, Ohio).

The preparation, characterization, and performance of various types of pyrolytic carbon (PyC)-coated fuel particles are discussed. Most of the coatings studied have been deposited by pyrolysis of either methane or acetylene in a fluidized bed of the fuel particles at temperatures between 950 and 2000°C. However, the preparation of

coatings by pyrolysis of other carbon compounds is also reported. Three types of coatings have been prepared: (1) monolithic—all PyC, (2) duplex—layers of porous and dense PyC, and (3) composite—layers of PyC and other materials. The out-of-pile characterization of these materials is presented along with a brief summary of their in-pile performance. All three types of coating have shown good out-of-pile integrity, but the duplex and composite coatings have given superior in-pile performance. The coating procedures are described, and process variables are correlated with deposition rate and coating properties. (auth)

52.

2735 PYROLYTIC CARBON ON CERAMIC FUEL PARTICLES. R. L. Beatty, J. L. Cook, and F. L. Carlson, Jr. (Oak Ridge National Lab., Tenn.). Trans. Am. Nucl. Soc., 7: 425-30 (Nov. 1964).

53.

37747 (A/CONF.28/P/398) IRRADIATION BEHAVIOR OF COATED FUEL PARTICLES. H. Bildstein, A. Burtcher, P. Koss, and F. P. Viehboeck (Österreichische Studiengesellschaft für Atomenergie G.M.B.H., Seibersdorf, Austria). May 1964. 17p.

The behavior of coated fuel particles under reactor conditions was investigated in a series of capsule tests performed in the ASTRA reactor at thermal flux levels of 3 to 6×10^{13} n/cm² sec. The burnup achieved was about 10% at operating temperatures of 1200 to 1400°C. Two types of capsules were used. One employed a graphite insert which was thermally insulated against the stainless steel containment by a helium gap. The other type utilized a niobium insert thermally insulated by Al₂O₃ powder. Both inserts contained the particles in an annular gap. The fuel temperature was monitored by high temperature thermocouples located at both sides of the fueled annulus. With this arrangement, data on the thermal conductivity of a loose packing of coated particles were obtained. After dismantling the capsules in the hot cell, the burnup was determined by evaluation of cobalt dosimeters and by mass-spectrometric analysis of the uranium isotopes. This measurement was performed by using a high-temperature ion source in which the uranium ions were obtained by surface ionization after a volatilization by a halogen treatment. After a visual inspection and a metallographical examination, the integrity of the particles was checked by two methods. The first utilized a xenon release measurement on reactivated particles to obtain the number of failed particles. The second used a chlorination process which allows the quantitative removal of uranium from failed particles via the volatile chloride. By analyzing the insert of the capsule, the amount of nongaseous fission products released during the experiments was determined by radiochemical methods and, in some cases, by gamma spectrometry. The results of the high temperature burnup tests are given and correlated with the results of a preirradiation evaluation of coated particles produced by different methods. (auth)

54.

39949 PROPERTIES OF CERAMIC-COATED NUCLEAR-FUEL PARTICLES. J. M. Blocher, Jr., M. F. Browning, W. J. Wilson, V. M. Secrest, A. C. Secrest, R. B. Landrigan, and J. H. Oxley (Battelle Memorial Inst., Columbus, Ohio). Nucl. Sci. Eng., 20: 153-70 (Oct. 1964).

Of the several factors tending toward failure of fuel-particle coatings in irradiation, fuel swelling, and radiation damage to the first 15–20 μm of coating by fission recoils are concluded to deserve the greatest immediate attention. Whereas ceramic coatings of sufficient thickness might withstand the stress, thinner coatings containing cushion layers or voids can be designed in principle to accommodate the effects of radiation damage. This was demonstrated for pyrolytic carbon and alumina coatings formed by chemical vapor deposition reactions. These principles cannot be applied to oriented coatings whose grains suffer anisotropic radiation damage by neutrons. This factor may ultimately limit the applicability of BeO coatings. Crushing strength, density, and hardness data are presented which show that pyrolytic carbon coatings with the same microstructure can have a wide range of properties. It is concluded that pyrolytic carbon coatings are not properly characterized by the terms laminar and columnar frequently used to describe their appearance. In the case of Al and Be coatings obtained by hydrolysis of the respective chloride vapors, deposition temperature is the most significant factor determining grain structure and permeability. Al₂O₃ coatings range from porous amorphous varieties, through an impermeable glass to elongated grains extending through the coating. The equivalent of the Al₂O₃ glass was not observed in the BeO coatings. MgO coatings can be expected to be of limited utility because of low strength. ZrO₂ coatings have the deficiency of permitting rapid oxygen diffusion. (auth)

55.

13676 (CA-5606(Pt 1)) ADVANCED GRAPHITE-MATRIX, DISPERSION-TYPE FUEL SYSTEMS AND RESEARCH ON GRAPHITE. Quarterly Progress Report for the Period Ending August 14, 1964. J. C. Bokros, W. V. Goettel, H. K. Lonsdale, R. E. Norman, R. J. Price, J. L. White, and L. R. Zumwalt (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Sept. 1, 1964. Contract AT(04-3)-167. 42p. Dep.(mn); \$2.00(cy), 1(mn) OTS.

Measurements of the mechanical properties of pyrolytic C deposited under various conditions show that the flexural modulus and the fracture stress decrease with increasing deposition temperature while the fracture strain increases, causing a net increase in elastic strain at fracture. The effect of annealing on the mechanical properties of pyrolytic C was studied. It was found that the largest changes in mechanical properties occur in material deposited at relatively low temperatures (1350 and 1450°C), where annealing causes a sharp decrease in modulus, a decrease in fracture stress, and a slight increase in fracture strain and that the changes in the mechanical properties of material deposited at high temperatures are relatively small. Seven representative structures of pyrolytic C were selected for tests of stability during exposure to high-energy neutron flux. The specimens were prepared and preliminary property measurements are being made. Graphite samples having various preferred orientations were prepared for irradiation testing in a capsule. Dimensional changes in the radial, longitudinal, and circumferential directions will be measured and compared with the dimensional effects predicted from preferred-orientation data. A previously derived relation between the preferred orientation and the Young's modulus of graphite bodies was tested for specimens of plastically deformed pyrolytic C.

Calculations of Young's modulus from measurements of the preferred orientation of these extremely anisotropic materials yield values in good agreement with those determined experimentally. In current studies of the hot-working characteristics of graphitic materials, bodies of high pitch content were hot-worked and the properties are being measured. Graphite specimens with various additives, e.g., NbC, ZrC, and Ni, were prepared to explore further the enhanced plasticity of carbide-containing graphite observed in previous work. Initial measurements of the tracer diffusion coefficients of Th in a laminar-type pyrolytic C were made; these values are significantly lower than those reported for U diffusion. (auth)

56.

7944 THORIUM-URANIUM CARBIDES FOR COATED-PARTICLE GRAPHITE FUELS. E. S. Bomar and R. J. Gray (Oak Ridge National Lab., Tenn.). Met. Soc., Am. Inst. Mining, Met. Petrol. Engrs., Inst. Metals Div., Spec. Rept., No. 13, 703-28(1964).

A description and pertinent results of research and development on thorium-uranium carbides for coated-particle, gas-cooled reactor graphite fuels are presented. The areas considered are as follows: pyrolytic-carbon-coating studies, in- and out-of-reactor evaluation of coated-particle materials, phase relationships in the Th-U-C system, microstructure of coated particles, impurity content of the particles, compatibility of carbides with pyrolytic carbon, and techniques for metallographic preparation and examination of reactive thorium-uranium carbides both before and after irradiation. Extensive investigation was made of the particle/coating interactions, as a function of time and temperature, and the effects of phase transformations on the performance of these materials under reactor conditions. An assessment of failure mechanisms indicated that the effects of particle swelling and fission recoils are of major significance. (auth)

57.

9469 THE COATING OF URANIUM DICARBIDE PARTICLES IN A FLUIDIZED BED. Carley-Macaulay, K. W. (Atomic Energy Research Establishment, Harwell, Eng.); Myles, J. W.; Williams, G. H. pp 826-48 of Carbides in Nuclear Energy. Vol. 2. Russell, L. E. (ed.). London, Macmillan and Co., Ltd., 1964.

The deposition of C and SiC coatings of varying thicknesses up to 100 μ , on UC_2 particles of about 200-400 μ dia is described. The coating was carried out in graphite vessels using a fluidization technique at temperatures between 1000 and 2000°C; developments are described in the design of these vessels for an increasing scale of operation. The structure and retention properties of the coatings were examined. It is concluded that preparation can be conducted on a reasonable scale in a fluidized bed reactor to make coatings which are effective as produced. Available in-pile tests on coated particles indicate that substantial retention of fission products may be maintained up to 1200-1300°C in-pile, in spite of considerable damage to the coatings. (auth)

58.

17351 (BMI-1659(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING DECEMBER 1963. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). Jan. 1, 1964. Contract W-7405-eng-92. 60p.

Research progress is reported on reactor materials and components, fuels, general fuel-element development, radioisotope and radiation applications, development of a tracer system for in-place filter evaluation, coated particle fuel materials, corrosion studies of the Fluoride Volatility Process, the USAEC-AECL cooperative program, fission product deposition studies, gas-cooled reactor program, and joining of regenerative cell components. (M.C.G.)

59.

20450 (BMI-1661(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING JANUARY 1964. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). Feb. 1, 1964. Contract W-7405-eng-92. 65p.

Progress is reported in programs relating to reactor materials and components, fuels, radioisotope and radiation applications, a tracer system for in-place filter evaluation, coated-particle fuel materials, corrosion in the fluoride volatility process, creep-rupture of SAP alloys, mechanical properties of Zr alloys, fission product deposition, UO_2 -ZrO₂, encapsulation, and joining of regenerative cell components. (D.L.C.)

60.

24968 (BMI-1665(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING FEBRUARY 1964. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). Mar. 1, 1964. Contract W-7405-eng-92. 59p.

Progress is reported on developments relating to civilian applications in reactor materials and components, fuel element fabrication and development, coatings for reactor fuels, fission product diffusion, radiation chemistry of organic compounds, and fuel burnup and feasibility. (N.W.R.)

61.

34032 (BMI-1668(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING MARCH 1964. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). Apr. 1, 1964. Contract W-7405-eng-92. 62p.

Progress is reported on reactor materials and components, fuel development, radioinduced formation of organotin compounds, coated-particle fuel development, corrosion studies in fluoride volatility process, creep-rupture of SAP alloys, mechanical properties of Zr alloys, fission product deposition, encapsulation studies, examination of irradiated Hastelloy X, and joining or regenerative cell components. (D.L.C.)

62.

34033 (BMI-1669(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING APRIL 1964. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). May 1, 1964. Contract W-7405-

eng-92. 59p.

Progress is reported on reactor materials and components, fuel development, radioinduced formation of organotin compounds, coated-particle fuels, corrosion studies in the fluoride volatility process, creep-rupture of SAP alloys, mechanical properties of Zr alloys, fission-product deposition, and irradiation of Hastelloy X and type 347 stainless steel. (D.L.C.)

63.

34034 (BMI-1672(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING MAY 1964. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). June 1, 1964. Contract W-7405-eng-92. 47p.

Progress is reported on: reactor materials and components, fuel development, solid-state pressure bonding, radioinduced formation of organotin compounds, coated-particle fuels, corrosion studies in fluoride volatility process, creep-rupture of SAP alloys, mechanical properties of Zr alloys, fission-product deposition, irradiation of Hastelloy X and type 347 stainless steel, and development of small radioisotope power sources. (D.L.C.)

64.

40970 (BMI-1674(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING JUNE 1964. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). July 1, 1964. Contract W-7405-eng-92. 46p.

Research and development progress on reactor materials and components, fuels, fuel elements, radioisotope and radiation applications, coated-particle fuel materials, materials development program, fuel irradiation, corrosion studies of the fluoride volatility process, creep-rupture of SAP alloys, mechanical properties of zirconium alloys, fission-product deposition, gas-cooled reactors, and small radioisotope power sources is reported. (M.C.G.)

65.

18420 (BMI-1694(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING SEPTEMBER 1964. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). Oct. 1, 1964. Contract W-7405-eng-92. 37p. Dep.; \$1.00(cy), 1(mn) CFSTI.

Progress is reported in research and development on reactor materials and components, reactor fuels, radioisotope and radiation applications, coated-particle fuel materials, fluoride-volatility process corrosive effects, and materials for ML-1 reactor. (J.R.D.)

66.

18421 (BMI-1697(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING OCTOBER 1964. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). Nov. 1, 1964. Contract W-7405-eng-92. 47p. Dep.; \$1.25(cy), 2(mn) CFSTI.

Research and development are reported on reactor materials and components, ceramic reactor fuels, radio-

induced formation of organotin, coated-particle reactor fuels, fluoride-volatility process corrosive effects, and gas-cooled reactor fuels. (J.R.D.)

67.

32029 (A/CONF.28/P/235) CERAMIC-COATED-PARTICLE NUCLEAR FUELS. R. W. Dayton (Battelle Memorial Inst., Columbus, Ohio), W. V. Goettel, and W. O. Harms. 14p.

Prepared for the United Nations Third International Conference on the Peaceful Uses of Atomic Energy, 1964.

A review is presented of the fabrication and performance of fuels consisting of coated uranium and thorium-uranium carbide or oxide particle dispersions in graphite or oxide matrices. The performance results indicate that pyrolytic carbon and alumina coatings retain fission gases, while alumina and beryllia coatings are nearly impermeable to all fission products. Two problems, the release of certain volatile fission products through pyrolytic carbon coatings and the deposition of strong beryllia coatings on uranium particles, are being studied. (D.L.C.)

68.

22844 (BMI-1704(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING NOVEMBER 1964. Russell W. Dayton and Stan J. Paprocki (Battelle Memorial Inst., Columbus, Ohio). Dec. 1, 1964. Contract W-7405-eng-92. 41p. Dep.; \$1.00(CFSTI).

Progress is reported on the development of reactor materials and components, on radioisotope and radiation applications, on coated-particle fuel materials, on corrosion in fluoride-volatility processes, and on gas-cooled reactor development. (W.L.G.)

69.

36042 (BMI-1679) FISSION-FRAGMENT-INDUCED DIMENSIONAL CHANGES IN PYROLYTIC CARBON. Thomas S. Elleman, Robert B. Price, and Duane N. Sunderman (Battelle Memorial Inst., Columbus, Ohio). July 23, 1964. Contract W-7405-eng-92. 17p.

To gain a better understanding of possible causes of coating failure of pyrolytic-carbon-coated fuel particles during irradiation, fission-fragment-induced dimensional changes were observed by: (1) lapping flat surfaces on coated fuel particles and measuring changes in the surface plane following irradiation, (2) measuring changes of the radius of curvature of unrestrained carbon tubes irradiated on the inner or outer surfaces with fission recoils, and (3) measuring the deflection of thin carbon disks after one face was irradiated with fission fragments. Fission-fragment-induced contractions were observed, but it was also possible to obtain behavior which appeared to indicate fission-fragment-induced expansion. (auth)

70.

40908 DEVELOPMENT AND UTILIZATION OF PYROLYTIC-CARBON-COATED CARBIDE FUEL FOR THE HIGH-TEMPERATURE GAS-COOLED REACTOR. Walter V. Goettel (General Atomic Div., General Dy-

namics Corp., San Diego, Calif.). Nucl. Sci. Eng., 20: 201-18(Oct. 1964). (GA-4699)

A brief review is presented of the development and evaluation of coated-particle fuel for the Peach Bottom high-temperature gas-cooled reactor. The results of extensive thermal and irradiation tests on the Peach Bottom fuel materials have shown them to be adequate for a 3-yr core life of this reactor. Consideration of the mechanisms that could lead to coating failure under irradiation has shown that swelling of the carbide particle from fission-product accumulation and shrinkage of the pyrolytic-carbon coating from fast-neutron and fission-recoil effects are the principal contributors to coating failure. A Triplex pyrolytic-carbon coating was developed to overcome these effects. This coating consists of a spongy layer of pyrolytic carbon overcoated with successive layers of laminar and columnar pyrolytic carbon. (auth)

71.

30115 (ORNL-3619(p.61-83)) FUELED-GRAPHITE DEVELOPMENT. A. Goldman (Oak Ridge National Lab., Tenn.).

The characterization of unsupported pyrolytic-carbon-coated UO_2 , ThO_2 , $(U,Th)O_2$, U carbide, and (U,Th) carbide fuel pellets as regards chemical properties, uniformity of shape, and mechanical properties is reported. Methods for eddy-current inspection and microradiography of coated fuel particles are outlined. Apparatus for infrared inspection of graphite spheres for defects is described. The fabrication of fueled graphite is discussed. The effectiveness of chlorination of fueled graphite for removal of externally and internally exposed fuel is discussed. An analysis is made of the chemical compatibility of pyrolytic carbon coatings on UO_2 , EuO_2 , and sol-gel $(U,Th)O_2$ pellets. The ThC_2 - UC_2 region of the Th-U-C phase diagram is studied. (T.F.H.)

72.

44054 (ORNL-3670(p.174-200)) GAS-COOLED REACTOR PROGRAM. A. E. Goldman and W. O. Harms (Oak Ridge National Lab., Tenn.).

The support of the Experimental Gas-Cooled Reactor includes receipt of the reactor fuel and the fabrication of instrumented fuel assemblies, assistance with welding problems, and measurements of stress-rupture properties of the fuel cladding material. The creep of the graphite moderator material under irradiation was found to help the material accommodate in-service strains and to be predictable by a theoretical model. Effects of irradiation on several physical properties of graphite were studied. A major effort is the development of unclad fuels for advanced gas-cooled reactors, especially graphite-base fuels containing dispersed pyrolytic-carbon-coated fuel particles. The properties of the coating were related to the conditions of deposition. The strain caused in the coating by irradiation was mathematically analyzed. An improved etching technique was found for preparing particles for metallography. Poison particles are also under study. Spherical graphite fuel elements have been tested

by abrasion, crushing, impact, thermal shock, radiography, eddy-current methods, and infrared emission, and prepared for reactor irradiation. The results have been related to design and fabrication. A phenolic resin was a better binder for graphite fuel elements than a furan. High-temperature chlorination removed exposed thorium and uranium from the surface of graphite fuel elements. Fueled beryllium oxide has been irradiated to 1.6% burnup of the fuel atoms. Uranium can be leached from fueled beryllium oxide with nitric acid. Supporting research for gas-cooled reactors has been in diverse areas. Using x-ray diffraction and other techniques, a phase diagram was prepared for the system UC_2 - ThC_2 . Methods were devised for welding new high-strength nickel alloys. Creep-rupture properties of a British stainless steel were measured under irradiation. The ratchetting of stainless steel cladding on ceramic fuel was studied, and it was found that the effect could be decreased by use of copper foil as a lubricant between the fuel and cladding. The mechanism of structural changes in UO_2 irradiated at high temperatures was investigated, and the thermal conductivity of graphite was measured from 50 to 1000°C. (auth)

73.

5117 (GA-5866) TARGET. A PROGRAM FOR A 1000-Mw(e) HIGH-TEMPERATURE GAS-COOLED REACTOR. Quarterly Progress Report for the Period Ending November 30, 1964. (General Atomic, San Diego, Calif. John Jay Hopkins Lab. for Pure and Applied Science). Dec. 31, 1964. Contract AT(04-3)-167. 139p. Dep. mn; CFSTI \$4.00 cy, \$1.00 mn.

Optimization programs were developed, and the fuel element development work progressed to the selection of the nonpurged packed-bed concept as the tentative reference design fuel elements. Irradiation and vibration tests of fuel elements were performed, as well as irradiation studies of carbon-coated fuel particles. Fission-product release and transport were stressed. Studies were made of fission product sorption and diffusion in graphite and of beryllium oxide-graphite compatibility. Preliminary investigations for the helium purification system were carried out including fission-product deposition and trapping studies and a study of hydrogen adsorption on activated carbon from high-pressure helium. Preliminary calculations of the gamma activities and the dose rates from ^{232}U daughter products in fuel were performed. Fuel cycle costs were analyzed, and methods for physics analyses were developed. A computer program for analyzing the prestressed concrete reactor structure was completed, and the liner for the first concrete model was tested. A theoretical investigation of thermal insulation for the TARGET prestressed concrete reactor structure was carried out. (D.C.W.)

74.

28998 (AERE-R-4260) REPORT OF A HIGH-TEMPERATURE REACTOR FUEL DISCUSSION GROUP. K. D. B. Johnson (United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England). Apr. 1964. 52p.

The results of a discussion group on requirements for future research and development in high-temperature reactor fuels, based mainly on coated fuel material, are

presented. Particular emphasis is placed on graphite-moderated systems having carbide-based fuel; oxides are considered in less detail. Phase stability, thermal conductivities, diffusion, and fission-product retention characteristics of several fuels are discussed. Uranium-thorium-graphite, uranium-thorium-carbon, pyrolytic graphites, uranium dicarbides, and carbide coatings are discussed. (R.E.U.)

75.

27864 (ORNL-3577) USE OF MICRORADIOGRAPHY COMBINED WITH METALLOGRAPHY FOR EVALUATION OF COATED PARTICLES. R. W. McClung, E. S. Bomar, and R. J. Gray (Oak Ridge National Lab., Tenn.). June 1964. Contract W-7405-eng-26. 16p.

The proposed use of nuclear fuels coated with pyrolytic carbon has made it necessary to devise techniques for evaluating them. Metallographic techniques, modified to accommodate the spherical particle configuration, have proved quite useful for observing the structure of both the cladding and the core. For nondestructive examination, special low-voltage radiographic techniques have been developed. By employing extremely fine grained photographic emulsions, optical magnification can be used to make very accurate measurements of the core diameter and coating thickness, and, in addition, the core and coating integrity can be evaluated. The combination of the methods permits the use of the best feature of each toward the evaluation of coated particles. (auth)

76.

30199 EVALUATING COATED PARTICLES OF NUCLEAR FUEL. Robert W. McClung, E. Sloan Bomar, and Robert J. Gray (Oak Ridge National Lab., Tenn.). Metal Progr., 86: No. 1, 90-3 (July 1964).

Metallographic and radiographic methods were combined and modified to evaluate coated fuel particles. Methods of rating structure, radiography exposure requirements, and microscopic examination of radiographs are discussed. Advantages and potentials of this procedure are outlined. (M.C.G.)

77.

2734 FLUIDIZED-BED CHARACTERISTICS IN THE COATING OF NUCLEAR FUEL PARTICLES. R. L. Piltoton and A. L. Lotts (Oak Ridge National Lab., Tenn.). Trans. Am. Nucl. Soc., 7: 429 (Nov. 1964).

78.

44743 (ORNL-P-1505) MATHEMATICAL MODEL FOR PREDICTING COATED-PARTICLE BEHAVIOR. Prados, J. W.; Scott, J. L. (Oak Ridge National Lab., Tenn.). [1964]. Contract W-7405-eng-26. 4p. (CONF-631101-18). Dep. mn; CFSTI \$1.00 cy, \$0.50 mn.

From 13th Conference on Remote Systems Technology. Washington, D. C.

A model was developed for predicting stresses and strains in pyrolytic-carbon coatings on fuel microspheres during irradiation. Calculations with this model lead to

the following predictions: coating failure may occur at the inner or outer surface, the former being more likely for enriched-uranium-based particles; a free volume of 2-3% of the fuel volume per at. % burnup ensures adequate coating life; isotropic outer coatings may be provided to prevent failure at the outer surface; increasing the outer-coating thickness beyond an inner dia/outer dia ratio of $\frac{2}{3}$ will not increase its life; and the use of low-density fuel particles should have little adverse effect on coated-particle life. (D.L.C.)

79.

14393 FUEL AND FERTILE MATERIALS. (Battelle Memorial Inst., Columbus, Ohio). Reactor Mater., 6: No. 4, 11-36 (Winter 1963-64).

Research and development progress is reported on uranium, α -uranium alloys, γ -uranium alloys, plutonium alloys, plutonium compounds, thorium, dispersion fuels, cermet fuels, coated-particle fuel materials, refractory fuel and fertile material, corrosion mechanisms for fuel alloys, and radiation effects on fuel materials. (83 references). (M.C.G.)

80.

18304 FUEL AND FERTILE MATERIALS. Reactor Mater., 7: No. 1, 14-35 (Spring 1964).

Research on unalloyed uranium, alpha-phase uranium alloys, gamma-phase uranium alloys, plutonium, plutonium compounds, thorium, dispersion fuels, cermet fuels, coated-particle fuel materials, uranium oxides, uranium carbides, uranium nitrides, uranium-sulfides, fuel-water reactions, and irradiation effects in fuel materials is reviewed. (70 references). (M.C.G.)

81.

40023 IRRADIATION STUDIES OF CERAMIC-COATED NUCLEAR FUEL PARTICLES. Charles W. Townley, Neil E. Miller, Robert L. Ritzman, and Richard J. Burian (Battelle Memorial Inst., Columbus, Ohio). Nucl. Sci. Eng., 20: 171-9 (Oct. 1964).

Irradiation studies of Al_2O_3 -, BeO -, and pyrolytic-carbon-coated fuel particles were carried out. Alumina-coated UO_2 particles were found to be capable of a high degree of fission-gas retention during irradiations to at least 10 per cent burnup at temperatures up to 1100°C. The use of thick Al_2O_3 coatings (about 60 microns) and porous UO_2 particles (about 80 per cent dense) was determined to be necessary to prevent cracking of the coatings at low temperatures. Coarse-grained beryllia coatings on UO_2 particles cracked during irradiations at 100°C and during thermal cycling in elevated temperature irradiations, but better performance is expected with fine-grained material. Failure of pyrolytic carbon coatings on UC_2 particles was prevented in low-temperature irradiations by using thick coatings (>100 microns), but at elevated temperatures it was learned that the coatings had to be multilayered as well. Very promising results were obtained for pyrolytic-carbon-coated UO_2 particles, good performance being observed over the temperature range of 100-1050°C. (auth)

82.

2727 (ORNL-P-166) TECHNIQUES FOR DEPOSITION AND SYSTEMATIC STUDY OF PROPERTIES OF PYROLYTIC CARBON ON CERAMIC FUEL PARTICLES. R. L. Beatty, J. L. Cook, and F. L. Carlsen, Jr. (Oak Ridge National Lab., Tenn.). [1963]. Contract W-7405-eng-26. 6p. (CONF-654-9). Dep.(mn); \$1.00(cy), 1(mn) OTS.

From American Nuclear Society Meeting, San Francisco, Nov.-Dec. 1964.

A brief description is given of the fluidized-bed coating of carbide and oxide fuel particles with pyrolytic carbon. The variables in the coating process are discussed, particularly the methane partial pressure in the case of UC_2 particles. The structure of the as-deposited coatings is discussed. (D.L.C.)

83.

30734 (EUR-1819.c(p.91-104)) THE INSTABILITY AND DIFFUSION PHENOMENA OF THERMAL TREATED PYROLYTIC CARBON COATED (U, Zr) C PARTICLES. F. Bencosky, P. Stecher (Metallwerk Plansee, Reutte/Tyrol (Austria)) and R. Thelien (European Atomic Energy Community, Ispra (Italy). Joint Nuclear Research Center).

The stabilization of UC against carbon at high temperatures by ZrC is discussed. Electron-beam microanalysis was carried out on pyrographite-coated UC-ZrC solid solutions after heat treatments. The observed U and Zr concentrations in the coated particles of varying composition are given. (D.L.C.)

84.

40168 (CONF-117) PREPARATION AND PROCESSING OF PARTICULATE NUCLEAR FUELS. Proceedings of a Symposium [Held at] New York, September, 9, 1963. Blocher, John M. Jr., (ed.) (Battelle Memorial Inst., Columbus, Ohio). Contract W-7405-eng-92. 202p. Dep. mn; CFSTI, \$2.00 cy, \$1.25 mn.

Twelve papers are presented on particulate fuels whose particle size, shape, distribution, or coating is a significant design factor in their end use. Requirements, fabrication, coating, and reprocessing are covered in the papers. Separate abstracts were prepared for five of the papers; for the other seven papers, see NSA 17: 25693; 18: 3875, 6785, 26008, 22520, and 33687; and 19: 20528. (D.L.C.)

85.

7902 (GA-4617(Pt.1)) ADVANCED, GRAPHITE-MATRIX, DISPERSION-TYPE FUEL SYSTEMS. Quarterly Progress Report for the Period Ending September 30, 1963. J. C. Bokros, W. V. Goettel, J. R. Hooker, H. K. Lonsdale, J. L. White, and L. R. Zumwalt (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Sept. 30, 1963. Contract AT(40-3)-167. 37p. Dep.(mn); \$2.00(cy), 1(mn) OTS.

Techniques have been developed for determining the density, crystallite size, layer spacing, and crystallographic orientation of pyrolytic carbon deposited in a fluidized bed. Characterization of pyrolytic carbons deposited over the range 1400 to 2100°C gave the following results: density was at a minimum in the 1600 to 1700°C range; the crystallite size increased uniformly with temperature of deposition; and preferred orientation data indicated that a highly oriented structure was deposited at 1400°C, random structures (almost isotropic) were pro-

duced at 1650°C (the density minimum), and slightly oriented structures were produced at temperatures above the density minimum. Other experiments indicated that variations in bed geometry and hydrocarbon partial pressure at constant temperature can also alter the structure of the deposits. Apparatus is being constructed with which to measure the mechanical properties of these pyrolytic carbons, so that the relationship between the structure of the deposits and their physical and mechanical properties may be determined. More complete data were obtained on the physical properties of graphite bodies (containing ZrC particles) hot-worked to moderate strain levels (approximately 30% in compression). Micrographic examinations of the hot-worked bodies indicated apparent improvements in microstructure as well as reduced porosity; however, no substantial improvement in permeability or in strength of the bodies was observed. The possibility that the samples were weakened during machining of the physical-property test specimens is being investigated. Removal of volatiles from pitch by either solvent extraction of pyrolysis continued to result in improvement in the properties of hot-pressed graphite compacts. Indications were that the material removed by volatilization during prepyrolysis was the same as that removed by petroleum-ether extraction, i.e., the molten fraction. The method used in the past to measure the bulk diffusion coefficients in the uranium-pyrolytic carbon system proved unsatisfactory because of edge effects and, presumably, macroscopic graphite imperfections. Measurements have been made that demonstrate the existence of significant surface migration in this system at temperatures as low as 1000°C. Experiments have been planned for the thorium-pyrolytic carbon system in which the recognized limitations of the previous method will be corrected. Fission-product release studies were performed on a series of fuel compacts that had been impregnated to various levels of permeability. No apparent correlation was found between the fission-product release and the permeability of the compacts. (auth)

86.

22455 (GA-4695) GRAPHITE-MATRIX FUELS DEVELOPMENT FOR THE AVR. Final Report. J. Chin, W. V. Goettel, J. R. Hooker, C. S. Luby, and D. Stevens (General Atomic Div., General Dynamics Corp., San Diego, Calif.). Oct. 1963. Contract AT(40-1)-3119. 74p.

Previously developed techniques for the preparation of radiation-stable coated particles and their fabrication into graphite-matrix spheres for pebble-bed reactors were applied to fuel preparation for the AVR. Triplex pyrolytic-carbon coatings consisting of an inner spongy layer, a middle laminar layer, and an outer columnar layer were applied to UC_2 and $(Th,U)C_2$ fuel particles. Several lots of these particles were supplied to ORNL for evaluation, including extensive irradiation testing. In addition, several types of graphite-matrix spheres containing Triplex particles were also fabricated and furnished to ORNL for testing. These fueled spheres (6 cm and 1 to $\frac{1}{2}$ in. in diameter) were initially prepared by hot-pressing techniques; in some, the coated particles were uniformly dispersed throughout the spheres, while others had an unfueled surface layer, the particles being dispersed in the central region (yolk) of the continuous matrix graphite. Techniques were also developed for fabricating fueled spheres using shells of machined graphite. (auth)

87.

30645 (EUR-1819.e(p.105-44)) THE APPLICATION OF X-RAY PROJECTION MICROSCOPY TO THE QUALITY CONTROL AND INSPECTION OF COATED PARTICLE FUELS FOR THE DRAGON REACTOR EXPERIMENT. J. F. G. Conde (European Atomic Energy Community, Ispra (Italy). Joint Nuclear Research Center).

After a description of the Dragon reactor fuel, the problem of assessing the coated fuel particles is considered. The x-ray projection microradiographic procedure which was established for the routine examination of Dragon coated-particle fuels is outlined, and the observed features are described. (D.L.C.)

88.

7985 (BMI-1650(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING SEPTEMBER 1963. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). Oct. 1, 1963. Contract W-7405-eng-92. 67p.

Progress is reported for projects in the following areas: reactor materials and components, fuel development, radiation applications in polymerizations, tracer system for in-place filter evaluation, development of coated-particle fuels, corrosion in fluoride-volatility process, mechanical properties of SAP and Zr alloys, fission product deposition, UO_2 -ZrO₂ studies, postirradiation examination of Hastelloy X cladding, and joining of regenerative cell components. (D.L.C.)

89.

15664 (BMI-1655(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING OCTOBER 1963. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). Nov. 1, 1963. Contract W-7405-eng-92. 77p.

Progress is reported on developments relating to civilian applications in reactor materials and components, studies of fuels, fuel element fabrication, coated-particle fuel materials, fission-product deposition studies, irradiation of materials and components, gas-cooled reactor program, joining of regenerative cell components, corrosion and oxidation studies, radioinduced polymerization and reactions, and fuel element burnup. (N.W.R.)

90.

13354 (BMI-1657(Del.)) PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING NOVEMBER 1963. Russell W. Dayton and Ronald F. Dickerson (Battelle Memorial Inst., Columbus, Ohio). Dec. 1, 1963. Contract W-7405-eng-92. 71p.

Research and development progress is reported on reactor materials and components, studies of fuels, general fuel element development, radioisotope and radiation applications, development of a tracer system for in-place filter evaluation, coated-particle fuel materials, corrosion studies of the fluoride-volatility process, progress on USAEC-AECL cooperative program, fission-product deposition studies, gas-cooled reactor program, and joining of regenerative cell components. (M.C.G.)

91.

4764 (ORNL-3470(p.129-56)) GAS-COOLED REACTOR PROGRAM. A. E. Goldman (Oak Ridge National Lab., Tenn.).

Approximately 65% of the first core for the EGCR and all control rods were received. Twelve of the twenty instrumented fuel assemblies for the EGCR core were assembled. A nondestructive technique was developed to detect nonbonding of the thermocouples to the cladding of these units. Preliminary results on graphite cantilever beams under irradiation at 450°F have shown that the zero intercept and the linear creep rate vary linearly with stress. Varied fuel elements and specimens were fabricated for irradiation testing for advanced reactor concepts. Suitable procedures were developed for arc welding end closures in beryllium tubes. A Ti-6 wt % Be alloy was found to have the best wetting behavior of brazing alloys for beryllium. The effects of external pressure, cladding thickness, and friction forces on interactions between UO_2 fuel and stainless steel cladding were studied. Pyrolytic-carbon-coated uranium carbide particles were evaluated by measurement of several physical characteristics. The escape of fission-product Kr^{88} from the particles appeared to be related to the percentage of particles cracked during load tests and the coating thicknesses. Equipment capable of coating particles at up to 2000°C has been assembled. Uranium migration in coated-particle fuels heated to 1600 and 2000°C was measured by neutron activation and by low-voltage radiography. Coated particles have retained fission gases excellently ($<10^{-4}\%$ release at 1400°C) when care was taken not to rupture coatings. Fueled-graphitic spheres and bushings were characterized. Dry chlorine gas at 800 to 1000°C markedly reduced exposed fuel, as measured by alpha counting, acid electrolysis, or neutron activation. Duplex pyrolytic-carbon coatings under irradiation have retained fission-product gases excellently. Only 1 to 2% of the coatings cracked on particles irradiated at 800 to 2100°F for up to 30 at.% burnup. Fueled graphite at 1500 to 1800°F irradiated to 2 at.% burnup retained Kr^{88} well (R/B $<10^{-7}\%$). Pyroxylin films have enabled measurement of the microhardness of pyrolytic-carbon coatings. These films retain the indenter impression; the pyrolytic-carbon coating is resilient and does not. Irradiation at 650°C has produced a greater effect on strength and ductility of the British 20 Cr-25 Ni-Nb alloys than has irradiation at 750°C. Irradiation appears to have weakened or embrittled the end-closure weld region. The metal cladding of UO_2 being irradiated appears to capture all energetic recoil atoms; the gases measured on postirradiation puncturing probably result from diffusion or from boiling out from the recoil tracks that penetrate the fuel surface. The thermal conductivity of polycrystalline UO_2 was found to fit a lattice-conduction heat-transport mechanism, at least to 1100°C. Deposition of fission-product iodine and tellurium from uranium carbide at 800°C was measured. The rate-controlling step in the deposition of I^{131} on stainless steel surfaces below 300°C appears to be diffusion through the helium sweep gas. Nitrided type 304 stainless steel was found to oxidize in impure helium at 1050°F. Thus under EGCR conditions the surface properties of the nitride case would be replaced by those of an oxide film. A

general interrelation was observed between strength, density, hardness, and modulus of elasticity of graphite moderator materials. The compression modulus of elasticity was much smaller than the corresponding tensile modulus for all specimens tested. (auth)

92.

2170 (ORNL-3512) INSTRUMENT FOR DETERMINING CRUSHING STRENGTH OF PARTICLES. George Hallerman and R. J. Gray (Oak Ridge National Lab., Tenn.). Nov. 29, 1963. Contract W-7405-eng-26. 14p.

An instrument for crushing-strength determinations of uncoated and pyrolytic-carbon-coated fuel particles (50 to 500 μ in diameter) was developed to relate the crushing strength of the particles to their fabricability. The instrument consists of a loading mechanism, load cell, and a power supply-readout unit. The information that can be obtained by statistical methods of the data analysis is illustrated by results on two batches of fuel particles. (auth)

93.

20463 (ORNL-TM-431) COATED-PARTICLE FUEL DEVELOPMENT AT OAK RIDGE NATIONAL LABORATORY. W. O. Harms, comp. (Oak Ridge National Lab., Tenn.). Feb. 14, 1963. Contract W-7405-eng-26. 75p. Dep.(mn); \$3.00(cy), 2(mn) CFSTI.

The research program on coated-particle nuclear fuel materials has been concerned with three general problem areas: (1) A broad evaluation of materials involving pyrolytic-carbon-coated fuel carbides with major emphasis on the relationship between measurable properties and behavior during irradiation under simulated gas-cooled power reactor conditions; (2) studies on the compatibility of coated-particle materials with liquid coolants; and (3) chemical processing development for graphite-matrix fuels containing coated particles. The techniques used and the significant results and implications that have evolved in this program are described. (auth)

94.

1263 FUNDAMENTAL ASPECTS OF THE DEVELOPMENT AND BEHAVIOR OF COATED-PARTICLE FUEL FOR THE DRAGON REACTOR EXPERIMENT. R. A. U. Huckle (United Kingdom Atomic Energy Authority, Winfrith, Dorset, Eng.). Trans. Am. Nucl. Soc., 6: 352(Nov. 1963).

95.

3155 PREPARATION AND TESTING OF COATED-PARTICLES FUELS. R. E. Latta and C. C. Browne (General Electric Co., Cincinnati). Trans. Am. Nucl. Soc., 6: 379(Nov. 1963).

96.

23996 (ORNL-3650(p.1-110)) REACTOR DEVELOPMENT. (Oak Ridge National Lab., Tenn.)

Progress is reported on Experimental Gas-cooled Reactor, materials studies, irradiation tests and studies German pebble-bed reactor, fueled graphite program, fast gas-cooled reactors, molten salt reactor program, High-

flux Isotope Reactor, army reactors program, space reactor program, thorium utilization program, reactor shielding, reactor physics, reactor fuels and materials, reactor components, fuel processing, nuclear safety, and waste disposal. (J.R.D.)

97.

30733 (EUR-1819.e(p.63-89)) APPLICATIONS OF X-RAY MICROANALYSIS AND X-RAY PROJECTION MICRORADIOGRAPHY TO THE STUDY OF UNIRRADIATED AND IRRADIATED HIGH TEMPERATURE REACTOR FUELS. R. Thelsen (European Atomic Energy Community, Ispra (Italy). Joint Nuclear Research Center).

Migration studies were conducted on coated carbide fuel particles. The results show that, at temperatures below 1700°C, U penetration proceeds along crystallite boundaries and coating defects. An effective diffusion barrier can be produced by a short-time annealing at 2000-2400°C. (D.L.C.)

98.

18396 (ORNL-3764) STUDIES ON EFFECTS OF DEPOSITION CONDITIONS AND HEAT TREATMENT ON THE PROPERTIES OF PYROLYTIC CARBON. Ji Young Chang (Oak Ridge National Lab., Tenn.). Mar. 1965. Contract W-7405-eng-26. 90p. Dep.; \$3.00(cy), 2(mn) CFSTI.

Thesis submitted to Univ. of Tennessee, Knoxville.

A study was made of the deposition of pyrolytic carbon from methane gas on direct resistance-heated graphite substrates; the deposits were characterized by means of density, crystallite size, preferred orientation and magnetic susceptibility measurements, and by microstructure studies. The effect of deposition conditions and heat treatment were evaluated in terms of anisotropy factors derived either directly from experimental susceptibility data or indirectly through calculation from other measurements. Deposits up to one millimeter thick were formed over the temperature range 1400 to 2400°C with subsequent heat treatment conducted at 2620°C for times up to 360 minutes on material deposited at 2200°C. All deposits had a columnar, cone-shaped grain structure perpendicular to the deposition surface. The crystallite size, degree of preferred orientation, and density increased and the unit cell height decreased with increasing deposition temperature and heat-treatment time. Evaluation of a diamagnetic susceptibility anisotropy factor, as the ratio of the diamagnetic susceptibility in the direction perpendicular to the deposition plane to the value measured in the plane, showed an increase in the value of this ratio for pyrolytic carbon deposited at progressively higher temperatures and heat treated for longer times. When the value of this direct experimental factor was compared to several calculated factors from x-ray and single-crystal susceptibility values, it was found that the calculated values were too high in every case involving as-deposited material. This behavior was interpreted as indicating that the calculations should not be made, assuming that single-crystal values for the susceptibility are correct, but rather that values characteristic of the state of imperfection of the individual crystallites in the deposit should be used. Since these data were not available, a reverse calculation was made to determine the ratio of the diamagnetic susceptibility in the c-plane to that perpendicular to the c-plane of the crystallites of each deposit. This calculated ratio was considerably higher than calculated from data for the best single crystals but did decrease toward the single-crystal value as the deposition temperature was increased. (auth)

99.

25571 (N-66-16152) KINETICS OF GRAPHITIZATION. I. THE HIGH-TEMPERATURE STRUCTURAL TRANSFORMATION IN PYROLYTIC CARBONS. Fischbach, D. B. (Jet Propulsion Lab., Calif. Inst. of Tech., Pasadena). Feb. 1, 1966. Contract NAS7-100. 40p. (NASA-CR-69917; JPL-TR-32-532). CFSTI \$2.00 cy. \$0.50 mn.

Under heat treatment at high temperatures, pyrolytic carbons undergo an extensive structural transformation. Layer stacking order, crystallite size, and preferred orientation increase. The kinetics of this transformation was investigated for several pyrolytic carbons in the temperature range 2400-3000°C. The transformation was found to be thermally activated, with an effective activation energy of about 260 kcal/mole. The fundamental graphitization process appeared to be the same for all of the carbons studied, but there were differences in their detailed kinetic behavior associated with differences in as-deposited microstructure. Variations in both the rate and mode of transformation can be understood in terms of the depen-

dence of the graphitization process on apparent crystallite layer diameter. A simple model of the graphitization process consistent with these observations is discussed. The significance of these observations for engineering applications is considered briefly. (auth)

100.

6698 CYTOVOLUMETRIC OBSERVATIONS ON EHRLICH ADENOCARCINOMA SUBJECTED TO X IRRADIATION. I. MODIFICATIONS INDUCED IN CELLS OF THE 1st TYPE. Paolini, F. A.; Rodolico, R.; Bandirali, G. R. (Univ., Pavia, Italy). Boll. Soc. Ital. Biol. Sper., 42: 343-6 (Apr. 15, 1966). (In Italian).

Forty Swiss strain mice were injected with a 0.2-ml suspension of Ehrlich adenocarcinoma ascites cells and x-irradiated 7 to 12 days later. The dose varied from 300 to 3000 R and was delivered at one time to the muscle site of the growing solid tumor. Cytovolumetric studies were made separately on the three types of cells as classified by Homann (Z. Krebsforsch., 58: 511(1951-2)), changes in type 1 cells being described in this report. Irradiation with 200 R produced no change in the volume of the nucleolus, nucleus, or cytoplasm. Irradiation doses of 600 R only slightly increased nucleolar volume, but appreciably enlarged the nuclear and cytoplasmic volume. No further increase in nuclear volume was noted with a 1000 R dose, the volume being 32% above the control value. Cytoplasmic volume was increased 31%. Further increases in nuclear and cytoplasmic volume occurred after 3000 R irradiation. (BBB)

101.

25584 PREFERRED ORIENTATION PARAMETERS FOR PYROLYTIC CARBONS. Fischbach, D. B. (California Inst. of Tech., Pasadena). Contract NAS7-100. J. Appl. Phys., 37: 2202-3 (Apr. 1966).

It is demonstrated that a variety of analysis techniques and characteristics parameters used for measurement of preferred orientation in pyrolytic carbon are equivalent, and that a simple relation exists between them. (L.B.S.)