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Informal Report

ONWI/Sub/79/E511-01200-5

MASTER

Urananite Leaching:

Literature Survey

University of California

LOS ALAMOS SCIENTIFIC LABORATORY

Post Office Box 1663 Los Alamos, New Mexico 87545

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Informal Report
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Urananite Leaching: Literature Survey

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K. E. Williams

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URANANITE LEACHING:
LITERATURE SURVEY

by

G. F. Grisham, E. A. Bryant, and K. E. Williams

ABSTRACT

A literature survey was undertaken to provide background material for a series of experiments involving the interaction of spent uranium dioxide fuel with various environments. Notes and references pertaining to the basic properties of UO_2 as produced and after reactor exposure are presented. The use of computerized literature searches is illustrated with specific topics related to leaching experiments.

INTRODUCTION

Uranium dioxide has been in use as a nuclear fuel since the first self-sustaining nuclear chain reactor was operated at the University of Chicago on December 2, 1942. The inherent properties of UO_2 for dimensional stability (both thermal and irradiation), its high melting point, resistance to corrosion when in contact with reactor components and its relative ease and low cost of production in large quantities in a highly pure form have led to its use in the majority of commercial nuclear power plants now operating in the US. Several other desirable properties can literally be manufactured during its sintering into fuel pellets.¹⁻⁸

By the time the first US commercial nuclear plant, Shippingport Atomic Power Station, began operating in 1957, the basic problems and solution for manufacturing fuel pellets with desired characteristics had been reported. Concurrent with the research and development in fuel manufacture was an extensive program of investigation of irradiation effects on the fuel within a reactor environment. The most pressing early problems were morphological changes in the UO_2 sinter, gaseous fission product buildup and release, and fission product migration to reactor components that caused chemical deterioration of those components.⁹⁻¹⁸

Advances in overall reactor technology have permitted long residence times for fuel material in a core with corresponding higher burnup of the initial ^{235}U charge and buildup of fission products and actinides. Investigation of the properties of these altered fuels has required advanced techniques in physical, radiochemical and metallographic measurements.¹⁹⁻³⁵

Ultimately, spent fuel must be removed from a reactor core and disposed. If chemical reprocessing is selected, then the dissolution of the fuel material obliterates its production and reactor thermal history. If the spent fuel is to be dealt with in an unprocessed form, then one must be appreciative of its history in order to characterize the material, a necessary first step before evaluating its future behavior in any environment. This implies a broad awareness of conditions which alter the physical and chemical behavior of uranium dioxide containing several weight percent highly radioactive fission and neutron capture products.

Part I of this report consists of fragmentary notes and observations made while surveying a few of the many readily available monographs, panel reports,

conference proceedings, journal articles and government agency reports written about uranium dioxide nuclear fuel during the past 25 years. Part II reviews briefly the use of computerized literature searches for locating published results of research endeavours having perhaps more direct bearing on an individual project.

Part I. Literature Survey Notes

This literature survey was undertaken to provide a limited bibliography pertaining to basic properties of uranium dioxide nuclear fuel and their changes after exposure in light water power reactors. The intended users were a group of radiochemists investigating the behavior of spent uranium dioxide fuel exposed to various aqueous environments. The main topics of interest were:

1. Chemical and physical properties of the pure compound UO_2 .
2. Properties of the nuclear fuel as manufactured and after reactor use.
3. Amount and distribution of fission products in the fuel elements at the end of their useful lifetime in a power reactor.
4. Experimental techniques that had been used to examine such highly radioactive material.
5. Results of any other studies of leaching of spent UO_2 fuel.

A. Properties of UO_2

Uranium dioxide as a stoichiometric compound is nearly mythical. The system U-O is the operative quantity. At stoichiometry, the crystal structure of UO_2 is fluorite. The uranium atoms define a face-centered cubic lattice with oxygen in the spaces between. This system permits the dissolution of excess oxygen or uranium at elevated temperatures with the excess atoms occupying the interstitial sites. With increasing hyperstoichiometry, the compounds U_4O_9 and ultimately U_3O_8 are formed. The phase equilibria of the U-O system have been reported extensively.^{1,6,13,36,37}

As a nuclear fuel, UO_2 is generally manufactured by pressing and sintering a pure powder under stringent conditions of temperature and reducing atmosphere.²⁻⁵ The resultant product is a nearly stoichiometric UO_2 ceramic material whose properties are a result of its preparation history.⁶⁻⁸ Typically, this material is greater than 90% theoretical density and has a homogeneous grain size of a few microns. The melting point of the material is $\sim 2840^\circ\text{C}$ and is a function of its O/U atom ratio. Physical properties such as thermal conductivity, elastic

modulus, Poisson's ratio, specific heat capacity, etc. which have significant bearing on the suitability of the material as a reactor component have been exhaustively investigated.³⁸

Most US commercial power reactors use uranium enriched in ^{235}U to make up the UO_2 fuel (2-3 atom% enrichments are typical.) The smallest UO_2 fuel unit for a reactor core is a pellet 8-15 mm in diameter and 15-20 mm long. These pellets are stacked in tubes of stainless steel or zircalloy to form a fuel element or rod. Depending on the reactor design, the pellets may be dished at their ends, spaced along the length of the tube with inert material, or both, and the completed rod generally has additional internal support hardware, gas plenum, etc. Current fuel rod design for pressurized water reactors have fuel rods up to 4m long which encompass the full length of the core. Multiple rods and various control hardware are further assembled into bundles. These units then form the exchangeable assemblies for the reactor core. Reference (19) provides descriptions of "classic" world-wide power reactors developed and commissioned by 1971. A summary of all US reactor experience through June, 1978 can be found in reference (20).

The changes in physical properties with time, temperature, and radiation exposure in a power reactor environment have been studied in great detail.^{9,10,13,16,17,21} The morphological changes in the UO_2 sinter are largely thermal effects caused by intense thermal gradients (up to 10^4 degC/cm) generated in the fuel pellet. During initial reactor run-up, fuel material undergoes further sintering or "densification", and exhibits radial cracking that does not anneal at higher temperatures. As burnup continues, the changes in grain structure and porosity become more dramatic.^{11,12} (Nearly all the references cited have excellent photographic displays of these morphological changes. Appendix A also has a set of typical photomicrographs of fuel material similar to that being used in the LASL leaching experiment.) At high burnup, the fission and neutron capture product accumulation in the fuel can become 10-20 atom percent of the UO_2 , and this is a chemically significant impurity within the fuel structure.^{22,23}

B. Fission Products: Generation and Redistribution

From the viewpoint of reactor operation and safety, the more mobile fission products, such as the gaseous elements Kr and Xe and the readily diffusive species Cs, Te, and I, pose the more serious technological problems. Consequently, a very large portion of the studies devoted to fission product behavior

during irradiation has been devoted to gaseous products and fuel-cladding interactions with diffusing species.^{14-18,26,27}

From the viewpoint of studying spent uranium dioxide fuel, the residual reaction products and their location within the fuel element and the fuel pellet matrix are the questions of interest. Further, the chemical state of this rather complex system will largely determine how it interacts with its environment. In the following discussion, the term "fission products" will include both "true" fission products and the actinide products of neutron reactions with heavy metal elements in the fuel material.

The number of fission products generated in a UO_2 fuel element is a function of the initial isotopic composition of the uranium, the power history of the core, and the location of the element within the core. This quantity, frequently called the "source term", is that to which subsequent operations on the element involving its fission product burden are referred. The source term is amenable to computer calculations given sufficient detail of power history, neutron fluence and energy distribution, fission cross sections and chain yields, neutron capture cross sections for the fission products, etc.³⁹⁻⁴³ The decay product inventory in the fuel at some time after its removal from a core is an extension of such computations. Dissolution of the fuel and subsequent radiochemical and mass spectrometric measurements of selected nuclei can imply total "burn-up" and neutron exposure of the material.^{32,33,44}

The fission product redistribution in a fuel element has the added complexity of radial temperature gradients and axial temperature distribution to that of the generation considerations.^{14-16,26} In practice, one sees only the result of these thermal forces as "frozen" in the fuel material during its final cool-down prior to removal from the core. This is the system that will interact with out-of-core environments.

The observation of individual radioactive nuclide distributions can be made readily if their characteristic decay radiations are measurable by gamma-ray spectrometric techniques.^{23,31} If present, the isotopes ^{137}Cs , ^{134}Cs , ^{144}Ce + ^{144}Pr , ^{95}Zr + ^{95}Nb , and ^{103}Ru , for example, can be detected by axial gamma-ray scanning of the intact fuel element. Radial isotope distributions are conventionally performed on cross sections of fuel pellets. New techniques combining gamma-ray spectroscopy and computerized data reduction permit radial mapping without sectioning.³⁴ For nuclides with only α , β or soft x-ray radiations,

samples of the fuel are taken for destructive analysis. Such analyses involve dissolving the material, chemically separating the element of interest, preparation of a sample suitable for detecting its radiations, and measuring those radiations. Autoradiographs of sectioned material can also be made but they merely indicate areas of activity, not specific nuclide concentrations.

The chemical state in which the fission products may be found is more difficult to determine than the radionuclide distribution. The following five categories generally describe the fate of chemical elements in spent fuel:²²⁻²⁹

1. Gases: Kr and Xe are trapped in voids.*
2. Metallic inclusions: Mo, Ru, Tc, Rh, Pd are found as metallic mixtures in melted zones, the central "chimney" and associated with columnar grains.* The observed inclusions have ranged in size from 2-10 μm in diameter.
3. Oxide precipitates ("grey" phases): U, Zr, Ba, O comprise the precipitates found in melted zones (trace Ce, Cs, Sr, Nd, La): oxides of Ba, Ce, Nd and Zr are associated with peripheral equiaxed grains.*
4. Dissolved in fuel matrix: Zr, Ce, Y, La, and other rare earths.
5. Volatiles: Cs, Te, Se, I, Rb deposit in/on any "cold" region, for example spacers, voids, cladding interfaces, or other internal hardware.

The principal tool for measuring elemental distribution is the electron microprobe.^{25,29,35} Metallographic techniques are employed for sample preparation, and various microscopy techniques are used to identify and measure size and distribution of inclusions.

C. Studies of UO_2 Leaching

With few exceptions, workers in the field of unprocessed fuel waste management have noted the paucity of information relating to spent UO_2 fuel behaviour under leaching conditions. The techniques for conducting leaching experiments have been established and standardized for solidified processed wastes.⁴⁵⁻⁴⁹ Extension of these techniques to spent UO_2 fuel has been made by workers at Battelle Pacific Northwest Laboratories.^{50,51}

*The conventions for describing grains and voids in spent fuel are given in reference (11) for example.

In addition to systematic investigations of leaching, one can find other references to irradiated UO_2 interactions with aqueous environments. For example, as early as 1963, Berman reported the results of a series of successive acid leaches of irradiated fuel undertaken to determine relative fission product concentration in grain boundaries vs that in solid solution in uranium dioxide.⁵² While his conclusions were speculative regarding fission product migration, they did add to evidence of phase separation accumulating from early metallographic and x-ray diffraction studies.

A true geologic leach experiment has been in progress for nearly two billion years in the natural fission "Oklo" reactor discovered by French scientists in Gabon, Africa. A review of the publications relating to this phenomenon is given in reference (53).

One occasionally finds insight into the aqueous leachability of UO_2 at a very practical level. A British report⁵⁴, pointed toward the build-up of radioactive cesium in fuel storage ponds, provides information about relative fission product and actinide release from fuel rods after varying reactor exposure and central fuel temperature.

Part II. Computerized Literature SEARCH - The RECON System

The use of computers for literature searches can greatly enhance one's grasp of the large body of publications relevant to the research and development which has been performed on uranium dioxide as a nuclear fuel. At the Los Alamos Scientific Laboratory Reference Library, the ERDA/RECON system is available for searches of several literature data bases, among them:

1. EDB-Energy Data Base (entries from 1976 to date)
2. NSA-Nuclear Science Abstracts (entries through May 1976)

The mechanics of using the RECON system are described in the ERDA/RECON Users Manual and the EDB Subject Thesaurus.^{55,56} Figures 1 and 2 illustrate RECON operations on the EDB and NSA files respectively (file sizes for EDB constantly increase as more citations are added; the NSA files are complete). Physical access to the system is generally through a keyboard terminal and dataphone link to the Oak Ridge National Laboratory where the computer and files reside. One selects the data base to be scanned and a set of "descriptors" (topics, authors, corporations, etc.) to be used for the scan (ref 56). The number of citations under major topics, such as URANIUM DIOXIDE and FUEL ELEMENTS, may be several

thousand. Thus, techniques for getting appropriate subsets must be employed. For example, the combining of these two sets will produce the file of entries to which both topics are referred (set 7 in Figs. 1 and 2). Such lists can be further limited to periods of time, type of publication, combined with other topics to further narrow the field of interest, etc.

Additional examples of limited subsets are also shown in Figs. 1 and 2. Set 9 in both examples contains the document references to URANIUM DIOXIDE and STOICHIOMETRY or OXIDATION. In both examples, set 10 is that list limited to journal publications only. Sets limited in time are shown in Fig. 1 set 11; entries in set 7 were limited to those made to the EDB data base during the period 1977-1978 (not necessarily the publication date). Sets 22 and 23, Fig 2, are NSA time-limited sets; for this data base, the limit is made on the volume numbers for the years in questions.

Having defined the files of interest, their contents can be obtained via a CRT display, an "on-line" printer at the keyboard terminal, or lists printed at ORNL and dispatched to the user. The entries in the file will have been sorted in reverse chronological order. In general, the length of the file will determine the output mode. The display of the citations is determined by the format chosen for that purpose. The complexity of the output ranges from a simple list of the citation numbers; number and title only; through number, title, source, abstract if available and "key words" contained within the publication. It should be noted that, in general, abstracts are available only in the EDB.

The printed output requested for both the EDB and NSA sessions are shown in Tables I and II. An index to each table has been provided to define again the contents of the sets, the formats for their display and any limits imposed. For the sake of brevity, only a few of most recent entries for each set are shown.

It should be noted that many libraries will perform the RECON operations with the requestor's guidance concerning subject material to be searched. Finally, selected libraries will also perform periodic up-dates to such searches on a subscription basis (for example; Western Regional Information Service Center, Lawrence Berkeley Laboratory, University of California, Berkeley, California).

Fig. 1. Typical RECON session history using the Energy Data Base.

Begin Session 006 -- File EDB was Selected

SET HISTORY

SET DESCRIPTOR	No. of CITS.
1. IT = Uranium Dioxide	1341
2. IT = Fuel Elements	2412
3. IT = Solubility	1653
4. IT = Stoichiometry	198
5. IT = Oxidation	4074
6. IT = Leaching	914
7. 1*2	215
8. 1*(3+6)	19
9. 1*(4+5)	67
10. LIM 9/ALL/J	13

Print 07/3/001-200 Accepted

Print 08/5/001-019 Accepted

Print 10/5/001-013 Accepted

Fig. 2. Typical RECON session history using the Nuclear Science Abstract base.

Begin Session 007 -- File NSA was selected

SET HISTORY

SET DESCRIPTOR	No. of CITS.
1. IT = Uranium Dioxide	6235
2. IT = Fuel Elements	10413
3. IT = Solubility	2408
4. IT Stoichiometry	719
5. IT = Oxidation	4215
6. IT Leaching	780
7. 1*2	1657
8. 1*(3+6)	230
9. 1*(4+5)	450
10. LIM 9/ALL/J	171
11. LIM 8/ALL/J	59
- ---	--
- ---	--
21. LIM 7/30-33	194
22. LIM 7/33	36

Print 10/3/001-171 Accepted

Print 08/2/001-200 Accepted

Print 11/2/001-059 Accepted

Print 21/3/001-194 Accepted

Print 22/2/001-036 Accepted

Table I Index

1. EDB PRINT 07/3/001-200 Accepted page 1-15
Set 07: Uranium dioxide * Fuel elements
Format 3: Title only
Limit: Most recent 200 entries
2. EDB PRINT 08/5/001-019 Accepted pages 15-19
Set 08: Uranium dioxide * (Solubility + Leaching)
Format 5: Title, source, abstract if available
Limit: All entries
3. EDB PRINT 10/5/001-013 Accepted pages 20-23
Set 10: Uranium dioxide * (Stoichiometry + Oxidation)
Format 5: Title, source, abstract if available
Limit: All journal entries

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78Y0113441
MIXED OXIDE FUEL FABRICATION/

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78P0108618
NUCLEAR FUEL ELEMENT/

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78C0108617
WELDING IN THE FABRICATION OF FUEL ELEMENTS AND NUCLEAR REACTOR COMPONENTS/

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RELEASE OF FISSION GASES FROM OXIDE FUELS (LWBR DEVELOPMENT PROGRAM)/

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EXAMINATION OF A FAILED ROD OPERATING WITH MOLTEN UO/SUB 2/--ZRO/SUB 2/--CAO FUEL (LWBR DEVELOPMENT PROGRAM)/

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BEHAVIOR OF GASEOUS FISSION PRODUCTS IN OXIDE FUEL ELEMENTS (LWBR DEVELOPMENT PROGRAM)/

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EFFECTS OF THERMAL GRADIENTS IN OXIDE FUELS (LWBR DEVELOPMENT PROGRAM)/

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78R0108263 (WAPD-TM--601)
MEASURED NATURAL NEUTRON SOURCE IN /SUP 233/UO/SUB 2/--ZRO/SUB 2/AND /SUP 233/UO/SUB 2/--THO/SUB 2/(LWBR DEVELOPMENT PROGRAM)/

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78R0108262 (WAPD-TM--596)
EVALUATION OF THE IRRADIATION BEHAVIOR OF A ZIRCALOY-4 CLAD ROD CONTAINING LOW DENSITY UO/SUB 2/FUEL PELLETS (LWBR DEVELOPMENT PROGRAM)/

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78R0108261 (WAPD-TM--595)
EVALUATION OF THE IRRADIATION PERFORMANCE OF ZIRCALOY-4 CLAD TEST ROD CONTAINING ANNULAR UO/SUB 2/FUEL PELLETS (ROD 79-19)(LWBR DEVELOPMENT PROGRAM)/

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78X0108244 (LA--7266-PR)
EXAMINATION OF FAST REACTOR FUELS, FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS, AND ANALYTICAL METHODS DEVELOPMENT: IRRADIATION TESTS. PROGRESS REPORT, JANUARY 1--MARCH 31, 1978/

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78X0107504 (HW--47050)
FUELS DEVELOPMENT OPERATION QUARTERLY PROGRESS REPORT, OCTOBER, NOVEMBER, DECEMBER 1956/

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78R0108098 (NUREG/CR--0073)
5/U ENRICHED UO/SUB 2/RODS IN WATER WITH FIXED NEUTRON POISONS/

OF 4.29 WT PERCENT /SUP 2

76R0052436 (DOCKET-50155--478)

BIG ROCK POINT NUCLEAR PLANT LICENSE NO. DPR-6, TECHNICAL SPECIFICATIONS, PROPOSED CHANGE, REVISION AND HEATUP ANALYSIS OF ALL-URANIUM G FUEL/

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ADVANCED LMFBR CORE DESIGN, SECOND QUARTERLY REPORT, OCTOBER--DECEMBER 1975/

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76R0052394 (ANL-CEN-RSD--76-1(SUPPL.1))

PROPERTIES FOR LMFBR SAFETY ANALYSIS/

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78J0075342 EDB-78-14 40.060

GAMMA RADIATION EFFECT ON SOLUBILITY RATE OF URANIUM OXIDES IN SODIUM CARBONATE SOLUTIONS/

GROMOV, V.V./PARKHAEVA, T.N. (AN SSSR, MOSCOW, INST. FIZICHESKOJ KHIMII)/

KHIM. VYS. ENERG./11/3/1977/

275-276/(IN RUSSIAN)

NONE/

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78J0069067 EDB-78-13 40.070

INTERACTION OF URANIUM(6) AND URANIUM(4) OXIDES WITH ZIRCONIUM PYROPHOSPHATE IN MELTS OF ALKALI METAL CHLORIDES, PRODUCTS

OF INTERACTION (URANIUM PHOSPHATES)/

KRYUKOVA, A.I./KORSHUNOV, I.A./KHRIPIKINA, Z.V./KAZANTSEV, G.N./SKIBA, O.V./

RADIOKHIMIYA/19/3/1977/

314-318/(IN RUSSIAN)

A STUDY HAS BEEN MADE OF PRODUCTS OF INTERACTION OF URANIUM OXIDES, UO₂/SUB 3/ AND UO₂/SUB 2/, WITH ZIRCONIUM PHOSPHATE IN THE MELT OF SODIUM AND POTASSIUM CHLORIDES AT 700/SUP 0/C. FORMATION OF MIXED PHOSPHATES OF THE MUO/SUB 2/PO/SUB 4/ AND MU/SUB 2/(PO/SUB 4)/SUB 3/ TYPES WAS FOUND TO TAKE PLACE AS A RESULT OF THE INTERACTION, WHERE M = NA, K. THESE COMPOUNDS WERE SYNTHESIZED. STUDY WAS MADE OF THEIR SOLUBILITY IN MELTS OF ALKALI METAL CHLORIDES AND OF THEIR STABILITY WITH RESPECT TO ABOVE MELTS AT 700-800/SUP 0/C./

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(ORNL/TH--6092) ADVANCED FUEL RECYCLE PROGRAM PROGRESS REPORT, JULY 1--SEPTEMBER 30, 1977/

UNGER, M.E. (COMP.)/

OAK RIDGE NATIONAL LAB., TENN. (USA)/

NOV 1977/at/

THIS QUARTERLY PROGRESS REPORT ON THE FUEL RECYCLE PROGRAM COVERS THE ONGOING DEVELOPMENT PROGRAM AIMED AT PROVIDING THE TECHNOLOGY FOR PRODUCTION-SCALE ADVANCED FUELS RECYCLE. THE CONTINUING BASIC LABORATORY AND HOT-CELL STUDIES, ENGINEERING RESEARCH, AND COMPONENT EQUIPMENT DEVELOPMENT ACTIVITIES ARE IN SUPPORT OF A COLD INTEGRATED EQUIPMENT TEST SCHEDULED FOR 1981 AND A CURRENT CONCEPTUAL DESIGN OF A HOT EXPERIMENTAL FACILITY SCHEDULED FOR OPERATION IN 1988./

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78C0042517 EDB-78-08 05.040

EXTRACTION OF METALS FROM ORES BY BACTERIAL LEACHING: PRESENT STATUS AND FUTURE PROSPECTS/

KELLY, D.P./

(UNIV. OF WARWICK, ENG.)/

PERGAMON PRESS INC./ELMSFORD, NY/1977/

MICROBIAL ENERGY CONVERSION/

SCHLEGEL, H.G./BARNEA, J. (EDS.)/

THE PRINCIPAL ORGANISM EFFECTING BACTERIAL LEACHING OF FERRUS AND SULFIDE ORES IS THIOBACILLUS FERROOXIDANS, THOUGH OTHER THIOBACILLI AND OTHER BACTERIA MAY BE INVOLVED. THE PROCESS DEPENDS ON (A) DIRECT SOLUBILIZATION OF METAL SULFIDES BY BACTERIAL OXIDATION (B) DISSOLUTION OF METAL SULFIDES OR OXIDES BY FERRIC IRON PRODUCED BY BACTERIAL PYRITE OXIDATION. MINING SPOIL DUMPS AND LOW GRADE ORES CAN BE LEACHED FOR COPPER OR URANIUM BY CHEAP LOW-LEVEL TECHNOLOGY. DUMP LEACHING ENABLES MAXIMUM RECOVERY OF VALUABLE METAL FROM ANY ORE, BUT MAKES POSSIBLE EXPLOITATION OF VERY LOW GRADE CU AND U ORES. CONTINUOUS EXTRACTION PROCESSES ARE POSSIBLE WHERE A CONTINUOUSLY GROWING BACTERIAL CULTURE IS FED WITH PYRITIC ORES (OR

9 MAY 1951/
QUARTERLY TECHNICAL REPORT/1 JAN-31 MAR 1951/

10/5/0000001-0000013// 1
78J0100731 EDB-70-20 22.090

RESPONSE OF UNIRRADIATED AND IRRADIATED PWR FUEL RODS TESTED UNDER POWER-COOLING-MISMATCH CONDITIONS/
MACDONALD, P.E./QUAPP, W.J./MEHNER, A.S./MARTINSON, Z.R./MCCARDELL, R.K./
NUCL.SAF./19/4/1970/
440-467/

THE REPORT SUMMARIZES THE RESULTS FROM THE SINGLE-ROD POWER-COOLING-MISMATCH (PCM) AND IRRADIATION EFFECTS (IE) TESTS CONDUCTED TO DATE IN THE POWER BURST FACILITY (PBF) AT THE U.S. DEPARTMENT OF ENERGY'S IDAHO NATIONAL ENGINEERING LABORATORY. THIS WORK WAS PERFORMED FOR THE U.S. NUCLEAR REGULATORY COMMISSION UNDER CONTRACT TO THE DEPARTMENT OF ENERGY. THESE TESTS ARE PART OF THE NUCLEAR REGULATORY COMMISSION'S FUEL BEHAVIOR PROGRAM, WHICH IS DESIGNED TO PROVIDE DATA FOR THE DEVELOPMENT AND VERIFICATION OF ANALYTICAL FUEL BEHAVIOR MODELS THAT ARE USED TO PREDICT FUEL RESPONSE TO ABNORMAL OR POSTULATED ACCIDENT CONDITIONS IN COMMERCIAL LIGHT-WATER REACTORS (LWRs). THE MECHANICAL, CHEMICAL, AND THERMAL RESPONSE OF BOTH PREVIOUSLY UNIRRADIATED AND PREVIOUSLY IRRADIATED LWR-TYPE FUEL RODS TESTED UNDER POWER-COOLING-MISMATCH CONDITIONS IS DISCUSSED. A BRIEF DESCRIPTION OF THE TEST DESIGNS IS PRESENTED. THE RESULTS OF THE PCM THERMAL-HYDRAULIC STUDIES ARE SUMMARIZED. PRIMARY EMPHASIS IS PLACED ON THE BEHAVIOR OF THE FUEL AND CLADDING DURING AND AFTER STABLE FILM BOILING./

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MIGRATION OF METALLIC FISSION PRODUCTS IN REACTOR OXIDE FUELS/
D'ANNUNCI, F. (EUROPEAN INST. FOR TRANSURANIUM ELEMENTS, (EURATOM) KARLSRUHE, GER.)/SARI, C./SCHUMACHER, G./
NUCL. TECHNOL./35/1/AUG 1977/
80-86/

THE FISSION-PRODUCT ELEMENTS MOLYBDENUM AND RUTHENIUM WERE ADDED TO URANIUM AND URANIUM-PLUTONIUM OXIDE BY A CO-PRECIPITATION TECHNIQUE. HEAT TREATMENT OF THESE MATERIALS IN A SIMULATED REACTOR THERMAL GRADIENT CAUSES MIGRATION AND COAGULATION OF THE METALS TO FORM INCLUSIONS UP TO A MAXIMUM SIZE OF 10 μ M. INCLUSIONS OF LARGE DIAMETERS BETWEEN 5 AND 10 μ M DO NOT MIGRATE NOTICEABLY BY DIFFUSION. THEIR MIGRATION RATE IS DIFFERENT FROM THAT OF PORES IN A TEMPERATURE GRADIENT./

10/5/0000001-0000013// 3
78J0051300 EDB-70-09 40.070

LOW-TEMPERATURE OXIDATION OF UO/SUB 2/, U/SUB 3/O/SUB 8/ AND U/SUB 3/O/SUB 8-X/ PHASE BY NITROGEN DIOXIDE/
TSVIGUNOV, A.N./KUZNETSOV, L.M./
RADIOKHIMIYA/10/3/MAY 1976/
411-413/(IN RUSSIAN)

IN THIS WORK THE PRODUCTS OF LOW-TEMPERATURE OXIDATION OF URANIUM DIOXIDE AND OCTAOXIDE AND THE PHASE U/SUB 3/ O/SUB (8-X) BY NITROGEN DIOXIDE WERE INVESTIGATED. IT WAS SHOWN THAT DEPENDING ON THE EXPERIMENTAL TEMPERATURE, STARTING MATERIALS, AND METHOD OF THEIR PRODUCTION, SEVERAL PHASES MAY BE SYNTHESIZED: EPSILON-UO/SUB 3/(I), EPSILON-UO/SUB 3/(II), OR ALPHA-UO/SUB 3/(III)./

10/5/0000001-0000013// 4
78J0020026 EDB-70-05 40.070

INFLUENCE OF ARGON-HYDROGEN ATMOSPHERE ON VITROCERAMICS STOICHIOMETRY/
BELEUTA, I.L. (INSTITUTUL DE FIZICA ATOMICA, BUCHAREST (ROMANIA))/
RADIOCHEM. RADIOANAL. LETT./20/1/1977/
37-43/

THE STOICHIOMETRY OF UO/SUB 2/ IN SEVERAL VITROCERAMICS OBTAINED BY MELTING IN DIFFERENT AR-H/SUB 2/ MIXTURES AT ABOUT 2300 DEG C WAS DETERMINED. THE MEASUREMENTS ARE IN GOOD AGREEMENT WITH THE STOICHIOMETRY DIAGRAM UO/SUB 2/SUB (X)-SiO/SUB 2 / CALCULATED FOR THE MELTS IN PURE HYDROGEN, BY USING $\Delta G/SUB 02/$, AS A FUNCTION OF MELTING CONDITIONS: PRESSURE, COMPOSITION OF THE MELT, TEMPERATURE. THE SiO/SUB 2 / LOSSES BY EVAPORATION ARE LOWER, AND THE SAMPLES CAST INTO FUSIBLE Moulds HAVE SMOOTHER SURFACES./

10/5/0000001-0000013// 5
78J0000180 EDB-70-02 21.030

STOICHIOMETRIC EFFECTS ON PERFORMANCE OF HIGH-TEMPERATURE GAS-COOLED REACTOR FUELS FROM THE U-C-O SYSTEM/

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EXAMINATION OF THE TOTAL PRESSURE OVER HYPOSTOICHIOMETRIC URANIUM DIOXIDE/
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HIGH TEMPERATURE OXIDATION OF URANIUM DIOXIDE.PART II.PRESSURE AND TEMPERATURE DEPENDENCE OF THE OXIDATION RATE/
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21J0006074

DIFFUSION UNDER ELECTRIC FIELD OF O AT TEMPERATURES FROM 900 TO 1000 DEG C IN UO/SUB 2./

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21J0004602

DISSOLUTION OF GRAPHITE-BASED REACTOR FUELS BY PRESSURIZED AQUEOUS PROCESSES./

8/2/0000001-0000200// 1

33J0030092 NSA-33-12 50.220

CARBOTHERMIC REDUCTION OF REFRACTORY METALS/

ANDERSON, R.N./PARLEE, N.A.D./

DEPARTMENT OF MATERIALS SCIENCE, SAN JOSE STATE UNIVERSITY, CALIFORNIA 95192/

J.VAC.SCI.TECHNOL./13/1/JAN 1976/526-529/

US/

US/

(JVSTA)

ACTIVATION ENERGY / ALUMINIUM OXIDES=M4 / BAUXITE / CARBON/CARBON MONOXIDE/CHEMICAL REACTION KINETICS/DECOMPOSITION/
 EXTRACTIVE METALLURGY/FORMATION FREE ENERGY/ILMENITE/IRON ORES/LIQUID METALS/MAGNESIUM OXIDES=M5/NITRIDES/ORE PROCESSING
 / PRECIPITATION /REDUCTION=Q1,Q2,Q3,Q4,Q5,Q6/SILICON OXIDES=M1/SOLUBILITY/SOLVENT EXTRACTION/THERMODYNAMICS/TIN/TITANIUM
 OXIDES=M3/URANIUM DIOXIDE=M6/URANIUM NITRIDES/URANIUM ORES/VAPOR PRESSURE/WETTABILITY/ZIRCONIUM OXIDES=M2/

8/2/0000001-0000200// 2

33J0022026 NSA-33-10 40.420

REACTION OF URANIUM DIOXIDE AND URANOUS/URANIC OXIDE WITH SODIUM MOLYBDATE/

PROKOSHIN, A.D./USTINOV, D.A./ANDRIANOV, H.A./CHEBOTAREV, N.T./LUKDYANOV, V.M./

SOV.RADIOCHEM.(ENGL. TRANSL.)/16/5/1974/632-634/

SU/

US/

(SVRDA)

CHEMICAL REACTIONS= Q1, Q2/MOLYBDATES/SODIUM COMPOUNDS/SODIUM URANATES=M3/SOLUBILITY=Q1,Q3/URANIUM DIOXIDE=M1/URANIUM
 OXIDES U3O8=M2/VERY HIGH TEMPERATURE/

8/2/0000001-0000200// 3

33C0022309 NSA-33-09 77.900

PREDICTION OF INERT-GAS SOLUBILITIES IN MOLTEN UO/SUB 2/ AND THEIR EFFECTS ON FUEL FRAGMENTATION AND BEHAVIOR/

GUNNERSON, F.S./CRONENBERG, A.W./

UNIV. OF NEW MEXICO, ALBUQUERQUE/

LMFBR/

TRANS.AM.NUCL.SOC./22/NOV 1975/409-410/

US/

US/

(TANSA)

AMERICAN NUCLEAR SOCIETY 1975 WINTER MEETING/

SAN FRANCISCO, CA, USA/

16 NOV 1975/

77.500/

FRAGMENTATION /FUEL-COOLANT INTERACTIONS=M,Q1/LMFBR TYPE REACTORS=M1/RARE GASES/SOLUBILITY/SORPTIVE PROPERTIES/URANIUM
 DIOXIDE/

8/2/0000001-0000200// 4

33R0011231 NSA-33-05 77.300

(KFK--2213) BEHAVIOR OF FISSION PRODUCTS IN HIGHLY BURNT HTR FUELS BY MICROPROBE ANALYSIS/

KLEYKAMP, H./

KERNFORSCHUNGSZENTRUM KARLSRUHE (F.R.GERMANY). INST. FUER MATERIAL- UND FESTKORPERFORSCHUNG/

OCT 1975/(IN GERMAN) DEP.NTIS (US SALES ONLY)\$4.00./

IC=3 586 000/

DE/

DE/

FUELS/IRRADIATION/NITRIC ACID/REPROCESSING/SOLIDS/SOLUBILITY/URANIUM DIOXIDE/

8/2/0000001-000200// 200

21R0043016 NSA-21-23 20.51

(GAMD--6360) FEASIBILITY STUDY FOR REPROCESSING HTGR FUELS BASED ON USING THE NUCLEAR FUEL SERVICES, INC., REPROCESSING PLANT./

REESE, S.L./BURN, P. (NUCLEAR FUEL SERVICES, INC., WEST VALLEY, N.Y.)/DONELSON, S.E./STEYER, K.G./

GENERAL DYNAMICS CORP., SAN DIEGO, CALIF/GENERAL ATOMIC DIV./

DEC. 15, 1964/

67P/DEP. CFSTI./

CHEMISTRY (SEPARATION)/

CC=611 7200/330 2000/

ABUNDANCE / CARBON / CHEMICALS / DESIGN/FLUORIDES/FUEL ELEMENTS/FUELS/GRAPHITE/HEATING/IRRADIATION/LEACHING/MACHINING/NEUTRON FLUX / POWER PLANTS / RADIOACTIVITY/REACTORS/RECOVERY/REPROCESSING/SAFETY/SEPARATION PROCESSES/SOLID SOLUTIONS/SOLUTIONS/SOLVENT EXTRACTION/STORAGE/TIME/URANIUM DIOXIDE/VOLATILITY/WASTE DISPOSAL/WASTE SOLUTIONS/WATER/THORIUM OXIDES/CONTAINMENT/

11/2/0000001-000059// 1

33J0030092 NSA-33-12 50.220

CARBOTHERMIC REDUCTION OF REFRACTORY METALS/

ANDERSON, R.N./PARLEE, N.A.O./

DEPARTMENT OF MATERIALS SCIENCE, SAN JOSE STATE UNIVERSITY, CALIFORNIA 95192/

J.VAC.SCI.TECHNOL./13/1/JAN 1976/526-529/

US/

US/

(JNSTA)

ACTIVATION ENERGY / ALUMINIUM OXIDES: M4 / BAUXITE / CARBON/CARBON MONOXIDE/CHEMICAL REACTION KINETICS/DECOMPOSITION/EXTRACTIVE METALLURGY/FORMATION FREE ENERGY/ILMENITE/IRON ORES/LIQUID METALS/MAGNESIUM OXIDES:M5/NITRIDES/ORE PROCESSING / PRECIPITATION /REDUCTION:Q1,Q2,Q3,Q4,Q5,Q6/SILICON OXIDES:M1/SOLUBILITY/SOLVENT EXTRACTION/THERMODYNAMICS/TIN/TITANIUM OXIDES:M3/URANIUM DIOXIDE:M6/URANIUM NITRIDES/URANIUM ORES/VAPOR PRESSURE/WETTABILITY/ZIRCONIUM OXIDES:M2/

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33J0022026 NSA-33-10 40.420

REACTION OF URANIUM DIOXIDE AND URANOUSURANIC OXIDE WITH SODIUM MOLYBDATE/

PROKOSHIN, A.D./USTINOV, D.A./ANDRIANOV, M.A./CHEBOTAREV, M.T./LUKOYANOV, V.M./

SOV.RADIOCHEM.(ENGL. TRANSL.)/16/5/1974/632-634/

SU/

US/

(SVRDA)

CHEMICAL REACTIONS: Q1, Q2/MOLYBDATES/SODIUM COMPOUNDS/SODIUM URANATES:M3/SOLUBILITY:Q1,Q3/URANIUM DIOXIDE:M1/URANIUM OXIDES U308:M2/VERY HIGH TEMPERATURE/

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32J0003917 NSA-32-02 50.130

A DILATOMETRIC STUDY OF THE SOLUBILITY OF U/SUB 4/O/SUB 9/IN UO/SUB 2//

BANNISTER, M.J./BUYKX, W.J. (AUSTRALIAN ATOMIC ENERGY COMMISSION RESEARCH ESTABLISHMENT, LUCAS HEIGHTS, MATERIALS DIV.)/

J.NUCL.MATER./55/3/MAR 1975/345-351/

AS/

NL/

(JNUMA)

COOLING / DILATOMETRY: Q3 /HEATING/HIGH TEMPERATURE/MEDIUM TEMPERATURE/PHASE TRANSFORMATIONS/SOLUBILITY:Q1/TEMPERATURE DEPENDENCE/THERMAL EXPANSION/TRANSITION TEMPERATURE/URANIUM DIOXIDE:M3/URANIUM OXIDES:M1/VERY HIGH TEMPERATURE/

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31J0009092 NSA-31-06 50.130

THE EFFECT OF TITANIUM ON GRAIN GROWTH AND DENSIFICATION OF SINTERED UO/SUB 2//

AINSCOUGH, J.B./RIGBY, F./OSBORN, S.C. (UKAEA REACTOR GROUP, SPRINGFIELD)/

J.NUCL.MATER./52/2/JCT 1974/191-203/

11/2/0000001-0000059//

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21J0004602 NSA-21-04 20-51

DISSOLUTION OF GRAPHITE-BASE REACTOR FUELS BY PRESSURIZED AQUEOUS PROCESSES./

HAAS, PAUL A./FERRIS, LESLIE M./

DAK RIDGE NATIONAL LAB., TENN./

IND. ENG. CHEM., PROCESS DESIGN DEVELOP., 5- 234-8 (JULY 1966)/

ORNL-P--1544/DAK RIDGE NATI/

COATING /CORROSION/FUELS/GRAPHITE/NITRIC ACID/OXIDATION/OXYGEN/PRESSURE/SOLUBILITY/VESSELS/WATER/URANIUM CARBIDES/(A;
THORIUM CARBIDES/(B) THORIUM OXIDES/(C) URANIUM DIOXIDE/

21/3/0000001-0000194//

1

33J0032532

IN-REACTOR MEASUREMENTS OF URANIUM DIOXIDE'S MODULUS OF ELASTICITY/

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2

33J0032280

BASIC PROPOSITIONS OF FORECASTING OPERATION CAPABILITY OF SODIUM-COOLED FAST REACTOR OXIDE FUEL ELEMENTS/

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ADVANCED LMFR CORE DESIGN, FIRST QUARTERLY REPORT, JULY--SEPTEMBER 1975/

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BEHAVIOR AND PHYSICO-CHEMICAL STATE OF FISSION PRODUCTS IN FUEL ELEMENTS FOR PRESSURIZED WATER REACTORS/

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REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, DECEMBER 1975/

21/3/0000001-0000194//

7

33X0025165 (ANL-RDP--45)

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1975/

21/3/0000001-0000194//

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33C0022509

STEADY-STATE CREEP MODEL FOR UO/SUB 2//

21/3/0000001-0000194//

9

33C0022508

IN-REACTOR RESTRUCTURING STUDIES OF UO/SUB 2//

21/3/0000001-0000194//

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33C0022397

THERMAL-SHOCK CRACKING IN UO/SUB 2/DURING POWER TRANSIENTS/

21/3/0000001-0000194//

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33C0022394

EXPERIMENTS ON PRESSURE-DRIVEN FUEL COMPACTION/

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30R0002018 (GEMP--600(VOL.3))
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THIRD ANNUAL REPORT:HIGH-TEMPERATURE MATERIALS AND REACTOR COMPONENTS DEVELOPMENT PROGRAMS.VOLUME II.MATERIALS/

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HIGH-TEMPERATURE MATERIALS AND REACTOR COMPONENT DEVELOPMENT PROGRAMS.VOLUME II.MATERIALS.FIRST ANNUAL REPORT/

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33J0032532 NSA-33-12 79.300
IN-REACTOR MEASUREMENTS OF URANIUM DIOXIDE'S MODULUS OF ELASTICITY/
BARANOV,V.M./BIBILASHVILI,YU.K./GOLOVNIH,I.S./KAKURIN,V.N./MEN'SHIKOVA,T.S.//
AT.ENERG.(USSR)/40/1/JAN 1976/37-40/(IN RUSSIAN)
SU/
SU/
(AENGA)
50.140/
ELASTICITY /FUEL ELEMENTS:M1/IN CORE INSTRUMENTS/IRRADIATION/PERFORMANCE TESTING/PHYSICAL RADIATION EFFECTS:Q2/REACTOR MATERIALS/ULTRASONIC TESTING:Q1/URANIUM DIOXIDE:M2/

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33J0032200 NSA-33-12 77.500
BASIC PROPOSITIONS OF FORECASTING OPERATION CAPABILITY OF SODIUM-COOLED FAST REACTOR OXIDE FUEL ELEMENTS/
GOLOVNIH,I.S./LIKHACHYOV,YU.I./
AT.ENERG.(USSR)/40/1/JAN 1976/27-37/(IN RUSSIAN)
SU/

SU/
(AENGA)
FUEL ELEMENTS=M2,Q1/LMFBR TYPE REACTORS=M1/PERFORMANCE=Q2/URANIUM DIOXIDE/

22/2/0000001-0000036// 3
33X0032275 NSA-33-12 77.500
(GEAP--14078-1)ADVANCED LMFBR CORE DESIGN.FIRST QUARTERLY REPORT,JULY--SEPTEMBER 1975/
GENERAL ELECTRIC CO.,SUNNYVALE,CALIF.(USA).FAST BREEDER REACTOR DEPT./
OCT 1975/AT/
IC=9 500 785/
US/
US/
COMPUTER CODES /COST/DESIGN=Q2/FUEL CYCLE/FUEL ELEMENTS/LMFBR TYPE REACTORS=M1/OPTIMIZATION/PLUTONIUM DIOXIDE/REACTOR
CORES=M2,Q1/REACTOR KINETICS/REACTOR MATERIALS/RESEARCH PROGRAMS/SPECIFICATIONS/URANIUM DIOXIDE/

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33J0032210 NSA-33-12 77.200
(NRC-TR-1)BEHAVIOR AND PHYSICO-CHEMICAL STATE OF FISSION PRODUCTS IN FUEL ELEMENTS FOR PRESSURIZED WATER REACTORS/
BAZIN,J./JOUAN,J./VIGNESDULT,M./
BULL.INF.SCI.TECH.(PARIS)/196/1974/55-71/DEP.MTIS \$3.50./
FR/
US/
(BUIAA)
FISSION PRODUCTS=Q2/FUEL ELEMENTS=M2,Q1/IRRADIATION/PERFORMANCE TESTING/PWR TYPE REACTORS=M1/URANIUM DIOXIDE/

22/2/0000001-0000036// 5
33C0028205 NSA-33-11 79.300
(IAEA--169(VOL.3))PROBLEMS ENCOUNTERED IN BURNUP DETERMINATION OF OXIDE FUEL SPECIMENS WITH A HIGH ROD POWER/
WERTENBACH,M./
INTERNATIONAL ATOMIC ENERGY AGENCY,VIENNA (AUSTRIA)/
1974/INIS/
FISSION PRODUCT NUCLEAR DATA (FPND).PROCEEDINGS OF A PANEL ORGANIZED BY THE IAEA AND HELD IN BOLOGNA,26-30 NOVEMBER
1973/
IC=3 294 000/
DE/
XA/
PANEL ON FISSION PRODUCT NUCLEAR DATA/
BOLOGNA,ITALY/
26 NOV 1973/
ACCURACY= Q2 / BURMJP / DIFFUSION / FAST FISSION / FISSION PRODUCTS/FISSION YIELD/FUEL ELEMENTS=M1/FUEL SCANNING=M2/
NONDESTRUCTIVE ANALYSIS=Q1/NUCLEAR FUELS/PLUTONIUM DIOXIDE/THERMAL FISSION/URANIUM DIOXIDE/

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33X0025166 NSA-33-10 77.500
(ANL-RDP--46)REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT,DECEMBER 1975/
ARGONNE NATIONAL LAB.,ILL.(USA)/
LMFBR/PUD/SUB 2/-UD/SUB 2//
1975/AT/
IC=0 440 000/
US/
US/
77.600/50.130/
(A) CLINCH RIVER BREEDER REACTOR=M11/(A)DESIGN=Q11/(A)REACTOR INTERNALS/(B)EBR-2 REACTOR=M10/(B)FUEL ELEMENT
FAILURE= Q10 / (B)REACTOR OPERATION=Q10/(B)TESTING/(C)GCFR TYPE REACTORS=M14/(C)RESEARCH PROGRAMS=Q14/(D)PLUTONIUM
DIOXIDE= M8/(D)STOICHIOMETRY/(D)THERMODYNAMIC PROPERTIES=Q8,Q9/(D)URANIUM DIOXIDE=M9/(E)FFTF REACTOR=M12/(E)REACTOR
OPERATION= Q12, Q13 /(E)TREAT REACTOR=M13/(F)COMPATIBILITY/(F)FUEL ELEMENTS=M4/(F)LMFBR TYPE REACTORS=M1/(F)REACTOR
COMPONENTS=M2/(F)REACTOR INSTRUMENTATION=M3/(F)REACTOR KINETICS=M5/(F)REACTOR MATERIALS=M7/(F)REACTOR SAFETY=M6/(F)
RESEARCH PROGRAMS= Q1,Q2,Q3,Q4,Q5,Q6,Q7/(F)SODIUM/(G)PROGRESS IS SUMMARIZED IN THE FOLLOWING RESEARCH AND DEVELOPMENT

APPENDIX
Photomicrographs of a Spent Fuel Element

The following photomicrographs have been included to fulfill the old adage "a picture is worth a thousand words". Fig. A-1 shows typical UO_2 powder prior to sintering. Figures A-3 through A-10 are photomicrographs of sections cut from a single fuel element at various positions along its length. These latter pictures were originally published in reference (57) and are reproduced here through the courtesy of P. E. MacDonald, EG&G, Idaho National Engineering Laboratory.

The leaching experiments under way at LASL involve sections of a spent fuel rod from the Carolina Power and Light H. B. Robinson 2 pressurized water reactor. Both rods, element K-4 shown in the photomicrographs and element E-14 being used at LASL, were part of assembly B05 and share a common irradiation and thermal history.

Information about element K-4 taken from reference (57) includes:

Pellet diameter	9.3 mm
Pellet Length	15.2 mm
Fuel stack Length	3.65 m
Density (% of theoretical)	$10.08 \times 10^3 \text{ kg/m}^3$ (92)
Fuel enrichment (wt %)	3.10
Average burnup (%)	2.84

A

3,000X

B

1,000X

C

300X

Fig. A-1. APS uranium dioxide powder, 7-8 μm diameter. (Courtesy K. V. Davidson, Los Alamos Scientific Laboratory).

Fig. A-2. Schematic diagram of the principal features shown in the fuel element photomicrographs ($\sim 8X$).

Fig. A-3. H. B. Robinson 2 fuel, assembly B05, element K-4, sample D-1. Transverse section 2.26 m from bottom of rod. Top view - initial section; bottom view - ground and polished to remove ~ 0.5 mm. (Courtesy P. E. MacDonald, EG&G, Idaho National Engineering Laboratory).

Fig. A-4. H. B. Robinson 2 fuel, assembly B05, element K-4, sample D-2. Transverse section 2.36 m from bottom of rod. Top view - initial section; bottom view - ground and polished to remove ~ 10 mm. (Courtesy P. E. MacDonald, EG&G, Idaho National Engineering Laboratory).

Fig. A-5. H. B. Robinson 2 fuel, assembly B05, element K-4, sample D-3. Transverse section 2.46 m from bottom of rod. Top view - initial section; bottom view - ground and polished to remove ~ 10 mm. (Courtesy P. E. MacDonald, EG&G, Idaho National Engineering Laboratory).

Fig. A-6. H. B. Robinson 2 fuel, assembly B05, element K-4, sample D-5. Transverse section 2.66 m from bottom of rod. Top view - initial section; bottom view - ground and polished to remove ~ 20 mm. (Courtesy P. E. MacDonald, EG&G, Idaho National Engineering Laboratory).

Fig. A-7. H. B. Robinson 2 fuel, assembly B05, element K-4, sample D-6. Longitudinal section, 2.87 to 2.89 m from bottom of rod. Top view - 0 to 180 degrees; bottom view - 180 to 360 degrees. Note gap between adjacent fuel pellets. (Courtesy P. E. MacDonald, EG&G, Idaho National Engineering Laboratory).

Fig. A-8. H. B. Robinson 2 fuel, assembly B05, element K-4, sample E-1. Transverse section 3.22 m from bottom of rod. Top view - initial section; bottom view - ground and polished to remove ≈ 0.2 mm. (Courtesy P. E. MacDonald, EG&G, Idaho National Engineering Laboratory).

Fig. A-9. H. B. Robinson 2 fuel, assembly B05, element K-4, sample E-2. Transverse section 3.27 m from bottom of rod. Top view - initial section; bottom view - ground and polished to remove ≈ 0.2 m. (Courtesy P. E. MacDonald, EG&G, Idaho National Engineering Laboratory).

Fig. A-10. H. B. Robinson 2 fuel, assembly B05, element K-4, sample E-5. Transverse section 3.52 m from bottom of rod. Top view - initial section; bottom view - ground and polished to remove ~ 10 mm. (Courtesy P. E. MacDonald, EG&G, Idaho National Engineering Laboratory).

REFERENCES

1. J. Belle, "Uranium Dioxide: Properties and Nuclear Applications," (U.S. Govt. Printing Office, Washington, D.C., 1961).
2. C. D. Harrington, "Preparation and Properties of Uranium Dioxide Powder," in Fuel Elements Conference, Paris, France, November 18-23, 1957 (USAEC Technical Information Service, Oak Ridge, Tennessee, 1958), TID-7546, Book 2, pp. 369-383.
3. L. C. Watson, "Production of Uranium Dioxide for Ceramic Fuels," in Fuel Elements Conference, Paris, France, November 18-23, 1957 (USAEC Technical Information Service, Oak Ridge, Tennessee, 1958), TID-7546, Book 2, pp. 384-401.
4. R. Kiessling and U. Runfors, "Sintering of Uranium Dioxide," in Fuel Elements Conference, Paris, France, November 18-23, 1957 (USAEC Technical Information Service, Oak Ridge Ridge, Tennessee, 1958), TID-7546, Book 2, pp. 402-413.
5. E. A. Evans and Staff, "Fabrication and Enclosure of Uranium Dioxide," in Fuel Elements Conference, Paris, France, November 18-23, 1957 (USAEC Technical Information Service, Oak Ridge, Tennessee, 1958), TID-7546, Book 2, pp. 414-431.
6. J. Belle and B. Lustman, "Properties of Uranium Dioxide," in Fuel Elements Conference, Paris, France, November 18-23, 1957 (USAEC Technical Information Service, Oak Ridge, Tennessee, 1958), TID-7546, Book 2, pp. 442-515.
7. J. E. Flinta, "Thermal Conductivity of Uranium Dioxide," in Fuel Elements Conference, Paris, France, November 18-23, 1957 (USAEC Technical Information Service, Oak Ridge, Tennessee, 1958), TID-7546, Book 2, pp. 516-525.
8. T. J. Heal, J. E. Littlechild, and R. H. Watson, "Development of Stable Density UO_2 Fuel," in Nuclear Fuel Performance, Proc. Int. Conf. British Nucl. Energy Soc., London, England, October 15-19, 1973 (Institution of Civil Engineers, London, 1973), Session 1, Paper 52, pp. 52.1-52.3
9. J. D. Eichenberg, P. W. Frank, T. J. Kisiel, B. Lustman, and K. H. Vogel, "Effects of Irradiation on Bulk Uranium Dioxide," in Fuel Elements Conference, Paris, France, November 18-23, 1957 (USAEC Technical Information Service, Oak Ridge, Tennessee, 1958), TID-7546, Book 2, pp. 616-716.
10. J. A. L. Robertson, "Some Consequences of Excess Oxygen in UO_2 ," in Symp. on Radiation Effects in Refractory Fuel Compounds, 64th Ann. Meet. Am. Soc. for Testing and Materials, June 27-28, 1961 (ASTM, 1962), Special Tech. Pub. No. 306, pp. 3-15.

11. J. L. Bates, "Voids in Irradiated UO_2 ," in Symp. on Radiation Effects in Refractory Fuel Compounds, 64th Ann. Meet. Am. Soc. for Testing and Materials, June 27-28, 1961 (ASTM, 1962), Special Tech. Pub. No. 306, pp. 16-28.
12. A. S. Bain, "The Heat Rating Required to Produce Central Melting in Various UO_2 Fuels," in Symp. on Radiation Effects in Refractory Fuel Compounds, 64th Ann. Meet. Am. Soc. for Testing and Materials, June 27-28, 1961 (ASTM, 1962), Special Tech. Pub. No. 306, pp. 30-46.
13. J. A. L. Robertson, "Irradiation Effects in Nuclear Fuels," an AEC Monograph (Gordon and Breach Science Publishers, 1969), Chap. 1, pp. 1-21 and Chap. 3, pp. 101-199.
14. J. A. Christensen, "Transport Processes in Oxide Nuclear Fuels," in Ceramic Nuclear Fuels, Int. Symp. Am. Ceramic Soc., Washington, D.C., May 3-8, 1969 (The Am. Ceramic Soc., Inc., 1969), pp. 109-125.
15. P. Chenebault and R. Delmas, "In Pile Mobility of Fission Gases in UO_2 Fuel Rods," in Ceramic Nuclear Fuels, Int. Symp. Am. Ceramic Soc., Washington, D.C., May 3-8, 1969 (The Am. Ceramic Soc., Inc., 1969) pp. 185-194.
16. B. R. T. Frost, "Studies of Irradiation Effects in Ceramic Fuels," in Ceramic Nuclear Fuels, Int. Symp. Am. Ceramic Soc., Washington, D.C., May 3-8, 1969 (The Am. Ceramic Soc., Inc. 1969), pp. 225-243.
17. D. A. Collins and R. Hargreaves, "Performance-limiting Phenomena in Irradiated UO_2 ," in Nuclear Fuel Performance, Proc. Int. Conf. British Nucl. Energy Soc., London, England, October 15-19, 1973 (Institution of Civil Engineers, London, 1973), Session 1, paper 50, pp. 50.1-50.5.
18. J. H. Davies and F. T. Ewart, "The Chemical Effects of Composition Changes in Irradiated Oxide Fuel Materials," J. Nucl. Mater., 41, 143-154 (1971).
19. M. T. Simnad, "Fuel Element Experience in Nuclear Power Reactors," an AEC Monograph (Gordon and Brach Science Publishers, New York, 1971), Sect. 2, pp. 181-430.
20. "Nuclear Reactors Built, Being Built or Planned in the United States as of June 30, 1978," (Technical Information Service Center, Oak Ridge, Tennessee, 1978), TID-8200-R38.
21. W. H. Zinn, W. P. Chernock, M. H. Bleiberg, H. M. Ferrai, T. Stean, M. F. Lyons, and G. J. Walke, "Development of Improved Light Water Reactor Fuels," Proc. Int. Conf. Peaceful Uses At. Energy, 4th, Geneva, September 6-16, 1971 (United Nations, New York, 1972), Paper A/CONF 49/P/080, Vol. 10, pp. 321-336.

22. C. E. Johnson, I. Johnson, P. Blackburn, J. E. Battles, and C. E. Crouthamel, "Stoichiometric Effects in Irradiated Uranium-Plutonium Oxide Fuel," in Behaviour and Ceramic State of Irradiated Ceramic Fuels, Proc. Panel Int. At. Energy Agency, Vienna, Austria, August 7-11, 1972 (IAEA, Vienna, 1974, STI/PUB/303), IAEA PL-463/1, pp. 1-30.
23. J. R. Findlay, "The Composition and Chemical State of Irradiated Oxide Reactor Fuel Material," in Behaviour and Ceramic State of Irradiated Ceramic Fuels, Proc. Panel Int. At. Energy, Agency, Vienna, Austria, August 7-11, 1972 (IAEA, Vienna, 1974, STI/PUB/303), IAEA-PL-463/2, pp. 31-40.
24. G. de Contensen, J. Monier, and N. Vignesoult, "Distribution des Produits de Fission et Localisation par Spectrometrie Gamma en cours d'Irradiation et après Irradiation," in Behaviour and Ceramic State of Irradiated Ceramic Fuels, Proc. Panel Int. At. Energy Agency, Vienna, Austria, August 7-11, 1972 (IAEA, Vienna, 1974, STI/PUB/303), IAEA-PL-463/9c, pp. 191-200.
25. J. Bazin, M. Perrot, and N. Vignesoult, "Analyse à la Microsonde de Produits de Fission Métallique dans les Combustibles en Géométrie Eau Ordinaire Fonctionnant avec une Zone Fondue," in Behaviour and Ceramic State of Irradiated Ceramic Fuels, Proc. Panel Int. At. Energy Agency, Vienna, Austria, August 7-11, 1972 (IAEA, Vienna, 1974, STI/PUB/303), IAEA-PL-463/9d, pp. 201-210.
26. J. R. Findlay, "The Migration of Fission Products Through Reactor Fuel Materials," in Behaviour and Ceramic State of Irradiated Ceramic Fuels, Proc. Panel Int. At. Energy Vienna, Austria, August 7-11, 1972 (IAEA, Vienna, 1974, STI/PUB/303), IAEA-PL-463/10 pp. 211-220.
27. J. Klíma, M. Poděšť, V. Vínš, "Contribution to the Studies of the Fission Gases Released from Irradiated Uranium Oxide Pellets," in Behaviour and Ceramic State of Irradiated Ceramic Fuels, Proc. Panel, Int. At. Energy Agency, Vienna, Austria, August 7-11, 1972 (IAEA, Vienna, 1974, STI/PUB/303), IAEA-PL-463/11, pp. 221-224.
28. P. Hoffmann and O. Götzmann, "Chemical Interactions of Fission Products with Stainless Steel Claddings," in Behaviour and Ceramic State of Irradiated Ceramic Fuels, Proc. Panel Int. At. Energy Agency, Vienna, Austria, August 7-11, 1972 (IAEA, Vienna, 1974, STI/PUB/303), IAEA-PL-463/14, pp. 237-254.
29. N. R. Stalica and C. A. Seils, "Electron Probe Microanalysis of Irradiated Oxide Fuels," in Ceramic Nuclear Fuels, Int. Symp. Am. Ceramic Soc., Washington, D.C., May 3-8, 1969 (The Am. Ceramic Soc., Inc. 1969) pp. 211-224.

30. A. C. Duckart, C. P. Ruiz, D. Kosanke and J. E. Menick, "Nuclear Fuel Rod Active Gamma Scan System," in Nuclear Fuel Performance, Proc. Int. Conf. British Nucl. Energy Soc., London, England October 15-19, 1973 (Institution of Civil Engineers, London, 1973), Session 8, Paper 70, pp. 70.1-70.4.
31. "Post-irradiation Examination Techniques," P. Cartledge, Ed., Proc. European Conf. British Nucl. Energy Soc., Reading, England, March 22-23, 1972 (Institution of Civil Engineers, London, 1972).
32. "Analytical Chemistry of Nuclear Fuels," Proc. of a Panel on the Anal. Chem. of Nucl. Fuels, Vienna, Austria, July 13-17, 1970 (International Atomic Energy Agency, Vienna, 1972).
33. "Analytical Chemistry in Nuclear Fuel Reprocessing," W. S. Lyon, Ed., Proc. 21st Conf. on Anal. Chem. in Energy Technology, Gatlinburg, Tennessee, October 4-6, 1977 (Science Press, 1978).
34. G. R. Waterbury, Los Alamos Scientific Laboratory, Gamma Scan Quality Assurance Procedure No's ANC-DE-1-GS-1, R3 and ANC-DE-1-GS-2, R2, unpublished work, 1979.
35. G. R. Waterbury, Los Alamos Scientific Laboratory, Electron-microprobe Quality Assurance Procedure No ANC-DE-1-MP-1, R3, unpublished work, 1979.
36. P. E. Potter, "Some Phase Equilibria and Thermodynamic Considerations for Irradiated Oxide Nuclear Fuels," in Behaviour and Ceramic State of Irradiated Ceramic Fuels, Proc. Panel Int. At. Energy Agency, Vienna, Austria, August 7-11, 1972 (IAEA, Vienna, 1974, STI/PUB/303), IAEA-PL-463/7, pp. 115-156.
37. T. M. Besmann and T. B. Lindemer, "Chemical Thermodynamics of the System Cs-U-Zr-H-I-O in the Light Water Reactor Fuel-Cladding Gap," Nucl. Tech., 40, 297-305 (1978).
38. P. E. MacDonald and L. B. Thompson, "MATPRO-Version 09, A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," EG&G Idaho, Inc. report TREE-NUREG-1005, (December 1976).
39. M. J. Bell, "ORIGEN-The ORNL Isotope Generation and Depletion Code," Oak Ridge National Laboratory report ORNL/TM-4628 (May 1978).
40. J. R. Marshall and J. A. Nowicke, "PLOTLIB-A Computerized Nuclear Waste Source-Term Library Storage and Retrieval System," Oak Ridge National Laboratory report K/OP-248 (March 1978).

41. T. R. England, W. B. Wilson and M. G. Stamatelatos, "Fission Product Data for Thermal Reactor-Part 1," Los Alamos Scientific Laboratory report LA-6745-MS (December 1976).
42. T. R. England, W. R. Wilson and M. G. Stamatelatos, "Fission Product Data for Thermal Reactors-Part 2," Los Alamos Scientific Laboratory, report LA-6476-MS (December 1976).
43. W. B. Wilson, T. R. England, and R. D. LaBauve, "Multi-group and Few-group Cross sections for ENDF/B-IV Fission Products; The TOAFEW Collapsing Code and Data File of 154-group Fission-Product Cross Sections," Los Alamos Scientific Laboratory report LA-7174-MS (March 1978).
44. W. J. Maeck, R. L. Tromp, F. A. Duce and W. A. Emel, "Isotope Correlation Studies Relative to High Enrichment Test Reactor Fuel," Idaho National Engineering Laboratory report ICP-1156 (June 1978).
45. E. D. Hespe, "Leach Testing of Immobilized Radioactive Waste Solids, A Proposal for a Standard Method," International Atomic Energy Agency, At. Energy Rev., 9 (1) 195-207 (1971).
46. J. E. Mendel, "A Review of Leaching Test Methods and the Leachability of Various Solid Media Containing Radioactive Wastes," Battelle Northwest Laboratory report BNWL-1765 (July 1973).
47. H. W. Godbee and D. S. Joy, "Assessment of the Loss of Radioactive Isotopes from Waste Solids to the Environment. Part I: Background and Theory," Oak Ridge National Laboratory report ORNL-TM-4333 (February 1974).
48. O. V. Anders, J. F. Bartel and S. J. Altschuler, "Determination of the Leachability of Solids," Anal. Chem., 50 (4), 564-569 (1978).
49. Klaus Becker, Draft ISO-Standard: Long-term Leach Testing of Radioactive Waste Solidification Products, International Organization for Standardization, Berlin, private communication, June 1978.
50. Y. B. Katayama, "Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Ground Water," Battelle Pacific Northwest Laboratories report BNWL-2057 (July 1976).
51. Y. B. Katayama and J. E. Mendel, "Leaching of Irradiated LWR Fuel Pellets in Deionized Water, Sea Brine, and Typical Ground Water," Battelle Pacific Northwest Laboratories report PNL-SA-6416 (December 1977).

52. R. M. Berman, "Fission Fragment Distribution in Irradiated UO_2 ", Nucl. Sci. Eng. 16, 315-328 (1963).
53. E. A. Bryant, G. A. Cowan, W. R. Daniels, and W. J. Maeck, "Oklo, an Experiment in Long-Term Geologic Storage," in Actinides in the Environment, ACS Symp. Series No. 35, New York, April 9, (American Chemical Society 1976), Paper 6, pp. 89-102.
54. D. A. Hilton, I. H. Robins and I. R. Brookes, "The Release of Water Soluble Species from Irradiated Uranium Dioxide," Berkely Nuclear Laboratories report CEGB-RD/N4106, Berkely, Gloucestershire, England (June 1977).
55. EDB Energy Information Data Base - Subject Thesaurus, TID-7000-R3 (June 1978).
56. ERDA/RECON Users Manual, TID-4586 (December 1976).
57. S. J. Dagbjartsson, B. A. Murdock, D. E. Owen and P. E. MacDonald, "Axial Gas Flow in Irradiated PWR Fuel Rods," Idaho National Engineering Laboratory report TREE-NUREG-1158 (September 1977).