HEDL-TI-76053

LITERATURE SEARCH ON LIGHT WATER REACTOR (LWR) FUEL AND ABSORBER ROD FABRICATION

1960 - 1976

Compiled by:

C. R. Sample, Reference Analyst
Battelle Pacific Northwest Library

for

M. J. Barr, HEDL Reactor Assembly

MASTER

This literature search was made in support of Task 7-LWR Recycle Fuel Study under the direction of Savannah River Laboratory.

February 1977

Hanford Engineering Development Laboratory

Operated by the Westinghouse Hanford Company

A Subsidiary of Westinghouse Electric Corporation

for the United States Energy Research and Development Administration

---

Blank Page

LITERATURE SEARCH ON LIGHT WATER REACTOR (LWR) FUEL AND ABSORBER ROD FABRICATION

Compiled by

C. R. Sample, Reference Analyst
Battelle Pacific Northwest Library

for

M. J. Barr, HEDL Reactor Assembly

ABSTRACT

A literature search was conducted to provide information supporting the design of a conceptual Light Water Reactor (LWR) Fuel Fabrication plant. Emphasis was placed on fuel processing and pin bundle fabrication, effects of fuel impurities and microstructure on performance and densification, quality assurance, absorber and poison rod fabrication, and fuel pin welding. All data have been taken from publicly available documents, journals, and books. This work was sponsored by the Finishing Processes - Mixed Oxide (MOX) Fuel Fabrication Studies program at HEDL.

Blank Page

CONTENTS

Blank Page

References

	Source of Citation
ANS	American Nuclear Society, Inc., 244 East Ogden Avenue, Hinsdale, Illinois, 60521.
Nuc. Tech.	Ibid.
ANS Trans.	Ibid.
Nuc. App and Tech (This title preceded ANS - American Nuclear Society)	Ibid.
Annals Nuc. En.	Incorporating Journal of Nuclear Energy, Peigamon Press, Headington Hill Hall, Oxford, England)X30BU (Oxford, 64881).
Atomindex	International Atomic Energy Agency, (Vienna, Austria).
"Meetings on Atomic Energy"	Ibid.
Journal of Applied physics	American Institute of Physics (New York, New York 10017).
Nuc. En. Mat.	Nuclear Energy Maturity Pergamon Press, New York, (Proceedings of the European Nuclear Conference, Paris, France).
NSA	Nuclear Science Abstracts, Unite States Energy Research and Development Administration (ERDA), Technical Information Center (Oak Ridge, Tennessee).
Phil. Mag.	Philosphical Magazine, Europhysics Journal, Taylor and Francis Ltd., 10-14 Macklin Street (London, England, WCZB5NF).
U.S. Patents	Official Gazette of the U.S. Patent Office (Washington, D.C. 20231).

I. FUEL AND FUEL ROD BUNDLE PROCESSING

Blank Page

3,917,768

SINTERED NUCLEAR FUEL AND METHOD OF PREPARING SAME

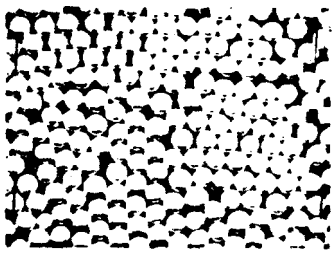
Giancarlo Abate-Daga, and Ignazio Amato, both of Turin, Italy, assignors to Fiat Societa per Azioni, Turin, Italy

Division of Ser. No. 885,524, Dec. 16, 1969, Pat. No. 3,759,786. This application Feb. 15, 1973, Ser. No. 332,835
Claims priority, application Italy, Feb. 25, 1969, 50722/69
Int. Cl.[2] G21C 21/00
U.S. Cl. 264—.5 5 Claims

1. A method of preparing a sintered nuclear fuel comprising a sintered mixture of uranium dioxide particles and consumable nuclear poison particles uniformly distributed therein, and wherein said poison is a rare earth oxide, said method comprising the steps of:

a. preparing microcapsules of the said poison of 10–2000 microns in diameter by gelling droplets of a sol of a hydroxide of the rare earth in a dehydrating liquid essentially consisting of a long-chain alcohol to form microcapsules, washing the gelled microspheres, heating the microspheres sufficiently to increase their density to at least 96% of the theoretical density, and then further heating the microcapsules to substantially the maximum temperature to be used in the sintering in step (e) hereof;

b. coating the microspheres obtained in step (a) with molybdenum metal by fluidizing said microspheres in an atmosphere consisting of an inert gas and molybdenum



hexacarbonyl vapor at an elevated temperature thereby to deposit molybdenum metal on the surface of the microspheres;

c. mixing the coated microspheres obtained in step (b) with ceramic grade uranium dioxide powder;

d. pressing the mixture of coated microcapsules and uranium dioxide powder into pellets; and

e. sintering the pellets under a hydrogen blanket.

U.S. PATENTS 1975

4829 (ORNL–4440, pp 192–215) DEVELOPMENT OF FUEL ELEMENT FABRICATION. Adamson, G. M. Jr. (Oak Ridge National Lab., Tenn.).

It was demonstrated that roll crushing UAl_x powders increases the yield of a usable product. However, the very high recoveries of about 90% that were previously reported are achieved only when crushed and remelted material is used; a single cycling of the material, with five remelts, yielded a recovery of only 75 to 80%. Variation in the concentration of U between 65 and 71 wt % had no effect on yield. The volume of voids in pressed and hot-rolled Al powder compacts, similar to the High Flux Isotope Reactor (HFIR) insert, was increased by the addition of 10 to 50 wt % ZrO_2 powder. The increase was less than found with fuel additions. Examinations by optical and transmission electron microscopy of type 6061 Al samples after cyclotron injection with 0.33 ppM H followed by various anneals revealed no evidence of blisters or bubbles. Blistering was encountered in plates in which both ends of the wrought Al cores were rounded in an attempt to reduce dogboning. Up to 9% void volume was obtained with plates fabricated for the HFIR outer annulus with 50 wt % loading of burned oxide fuel. Plates for tests of high burnup of a heavily loaded core were prepared and are awaiting insertion. As with the full-size plates, the highest void contents were obtained with the burned oxide fuel. No fabrication effects could be determined that were dependent upon fuel particle size, even with 100% –325 mesh powders. Examination of a second spent HFIR fuel element was completed except for post irradiation blister-annealing tests. In general, the element was found to be in excellent condition, with no indications of potential failure or significant dimensional changes. The oxide film on the surfaces of the fuel plates in this element was significantly thinner than that found on plates from the previous element, apparently because a thin oxide film was intentionally produced on the surfaces of the plates in the earlier HFIR fuel element before they were placed into operation. Studies on the measurement of fuel inhomogeneity by radiographic densitometry were completed. (auth)

NSA, vol. 24, 1970

29924 (ORNL–4630, pp 233–8) DEVELOPMENT OF FUEL ELEMENT FABRICATION. Adamson, G. M. Jr.; Martin, W. R. (Oak Ridge National Lab., Tenn.).

In Fuels and materials development program quarterly progress report for period ending September 30, 1970.

The irradiation experiment planned for insertion into the High Flux Isotope Reactor is being fabricated. The parametric studies of rolling shaped cores to evaluate nonuniform deformation (dogboning) have shown that small, transverse cracks are formed in the cladding over the dogbone region of a shaped core. These cracks are difficult to find after rolling unless the plates are etched. Assistance was provided in evaluating the future of fuel elements irradiated in the Advanced Test Reactor. Work results indicated that the large pimples were likely formed by an interaction of reactor coolant with core materials after the cladding had failed. The cause of cladding failure is unknown. (auth)

NSA, vol. 25, 1971

49507 (ORNL–TM–3300, pp 269–75) DEVELOPMENT OF FUEL ELEMENT FABRICATION. Adamson, G. M. Jr.; Martin, W. R. (Oak Ridge National Lab., Tenn.).

The irradiation experiment, designated PM capsule 1, was built and will be inserted into the HFIR to evaluate several properties that are believed to influence the performance of both ATR and HFIR fuel elements at high fission densities. (auth)

NSA, vol. 25, 1971

10256 (U-Pu)C PARTICULATE FUEL FABRICATION. Ayer, J. E. (Argonne National Lab., Ill.). Nucl. Met., Met. Soc. AIME, 13: 249-62(1968).

From the Nuclear Metallurgy Conference, Phoenix, Ariz. See CONF-671005.

A series of uranium-plutonium carbide elements were fabricated for fuel element performance studies. The fuel elements were manufactured by vibratory compaction employing an infiltration technique. The loading process parameters were determined on the basis of model studies carried out on nonfissile materials.

Eleven rods containing 80% uranium carbide and 20% plutonium carbide and five rods containing solid-solution carbides were fabricated. The mixed carbide fuel elements were all loaded to smear densities of 80% of theoretical with the exception of one which was compacted to 83.8% theoretical density. The solid-solution rods were formed to fuel densities of 84.2 ± 1.3% of theoretical. Fabrication revealed that the packing characteristics of carbides deviated from the model studies due to attrition of the fuel material. In some cases 20 percent of the first component addition was reduced to finer screen sizes during loading. Uniformity of density is promoted by infiltration, although angular particles are conducive to low infiltration rates. (auth)

NSA, vol. 23, 1969

27130 FABRICATION OF NUCLEAR FUEL RODS. Bairiot, H.; Aerts, L.; Chermanne, J. M. (to Belgonucleaire SA). Belgian Patent 729,682. 11 Mar 1969.

A process is described for the fabrication of a fuel rod for a nuclear reactor characterized by the fact that there is added to the interior of the can a compound that is nonvolatile at ordinary temperature but volatile under conditions of reactor operation and which is burned progressively during irradiation of the rod in the reactor. (BE)

NSA, vol. 27, 1973

21977 (CONF-750117-P1, pp 303-352) INTERRELATIONSHIPS BETWEEN NUCLEAR FUEL DESIGN, PERFORMANCE, AND FABRICATION. Bement, A. L. (Massachusetts Inst., of Tech., Cambridge). 1975.

From international symposium on nuclear power technology and economics; Taipei, China, Mainland (13 Jan 1975).

In international symposium on nuclear power technology and economics. Volume I. (Institute of Nuclear Energy Research, Lung-Tan (Taiwan); National Science Council, Taipei (Taiwan)).

Several factors which indicate that LWR nuclear fuel design and fabrication are still undergoing active technical evolution are described. Some aspects of design and fabrication technology where additional research and development are needed are discussed. The highly interactive nature of design and fabrication development is examined. (JWR)

NSA, vol. 33, 1976

16519 NUCLEAR FUEL ELEMENTS. Biancheria, Amilcard; Allio, Robert J. (to Westinghouse Electric Corp.). U. S. Patent 3,427,222. Feb. 11, 1969. Filed Oct. 15, 1965.

A nuclear fuel element is described which has a configuration of an elongated tube with closed ends. The tube contains ceramic bodies of sintered fissionable material; and a coating on the bodies which comprises a predetermined amount of an element functioning as a burnable poison. (Offic. Gaz.)

NSA, vol. 23, 1969

52653 FUEL ELEMENT FOR ATOMIC REACTORS. Bigge, W. B.; Roupe, G. A.; Lass, J. L.; Dominic, A. V. (to General Electric Co.). Norwegian Patent 166,011. 15 Dec 1966. Priority date 17 Jan 1966, United States. (In Norwegian).

In order to compensate for slight differences between the desired reactivity properties of a new fuel charge and the actual properties of the fuel as designed based on earlier data, it is proposed that a number of fuel pin locations along, and symmetrically perpendicular to, the main diagonal of a square lattice fuel assembly be arranged so that the fuel pins are easily removable. The reactivity may then be adjusted by replacing standard fuel pins by pins with higher or lower enrichment or by pins containing burnable poison. A locking system is shown wherein a locating collar on the upper grid plate has a slot and a recess to accommodate a pin on a spring-loaded sleeve on the top end of the fuel pin. This forms a kind of bayonet mechanism. A number of examples are given illustrating the applications of the invention. (Norway)

NSA, vol. 26, 1972

45957 NUCLEAR FUEL COMPRISING URANIUM DIOXIDE IN A POROUS CERAMIC OXIDE MATRIX. Burke, Thomas J.; Belle, Jack; Clayton, John C. (to United States Atomic Energy Commission). U. S. Patent 3,657,137. 18 Apr 1972. Filed 4 Jun 1964.

A new ceramic-type nuclear fuel consisting of particles of uranium dioxide embedded in a matrix of zirconium oxide, in which the uranium dioxide particles are partially separated from the matrix by an annular void is prepared by mixing uranium trioxide with calcium-oxide-containing zirconium dioxide powder and a small quantity of a binder, pressing the mixture, and firing the mixture in an atmosphere of hydrogen. (Official Gazette)

NSA, vol. 26, 1972

3,657,137

NUCLEAR FUEL COMPRISING URANIUM DIOXIDE IN A POROUS CERAMIC OXIDE MATRIX

Thomas J. Burke, Jack Belle, and John C. Clayton, Pittsburgh, Pa., assignors to the United States of America as represented by the United States Atomic Energy Commission

No Drawing. Filed June 4, 1964, Ser. No. 374,227
Int. Cl. C09k 1/30
U.S. Cl. 252—301.1 2 Claims

A new ceramic-type nuclear fuel consisting of particles of uranium dioxide embedded in a matrix of zirconium oxide wherein the uranium dioxide particles are partially separated from the matrix by an annular void is prepared by mixing uranium trioxide with calcium-oxide-containing zirconium dioxide powder and a small quantity of a binder, pressing the mixture and firing the mixture in an atmosphere of hydrogen.

U.S. PATENTS 1972

5048 (ORNL-4470, pp 94-111) FUEL ELEMENT FABRICATION DEVELOPMENT. Adamson, G. M. Jr.; Martin, W. R. (Oak Ridge National Lab., Tenn.).

Irradiation testing of UAl_x and burned U_3O_8 powders as dispersion fuels was extended to higher loadings and burnups. No evidence of potential failure was noted at conditions proposed for advanced HFIR cores. Significantly less swelling was noted for the burned U_3O_8 fuels than for high-fired U_3O_8 or UAl_x. The amount of swelling may be predicted from the initial void volume in the plates and the burnup. A second irradiated HFIR fuel element was examined in the hot cells. It was in better condition than the first one, with no significant dimensional changes or metallographic evidence of incipient failure. The oxide deposit was much more uniform and was thinner. Formation of the dogbone shape during rolling of fuel plates was shown by the use of high-speed motion pictures. During much of the rolling, voids are present at both ends of the core. Techniques for reduction of the amount of dogboning are proposed. The volume of fabrication voids in such plates depended on both the type and concentration of the U_3O_8. Conditions reducing dogboning also reduce the void volume. Sol-gel U_3O_8 offers excellent possibilities for increasing the void volume. The fit between the fuel cores and the frames before rolling was not as critical as previously expected. Miniature fuel plates electrolessly coated with nickel-phosphorus alloy have been placed within the reactor. On samples, the coating resisted spalling from thermal shock and treatment at 300°C but spalled completely after being held at 500°C. Anodic films were developed that protected pure aluminum against corrosion in static tests but not aluminum alloys under dynamic conditions with high heat fluxes. Suitable electron-beam welds were made with either type 6061 or 2219 aluminum in HFIR joint geometries. While adequate penetration was achieved, the strengths are lower than desired. Explosive welding techniques were developed for attaching spacers to HFIR fuel plates. While only intermittent bonds have been achieved in the preliminary tests, the technique appears to be adaptable but expensive. For applications below about 200°C, the tensile strength of dispersion-strengthened aluminum can be enhanced by dissolving magnesium. The room and high-temperature strengths and ductility of $Al-Al_2O_3$ dispersions can be improved by use of atomized alloy powders containing Fe, Mo, Zr, Ti, and Cr. (auth)

NSA, vol. 24, 1970

15212 FUEL AND FUEL CYCLE FOR OSKARSHAMN. Almgren, B. Nucl. Eng. Int.; 15: 992-5(Dec 1970).

The fuel cycles for Units I & II are discussed and compared. Details are given of U procurement and conversion of UF_6; toll enrichment; fuel assembly design; fuel fabrication; core layout and reactivity control; fuel performance; monitoring of power distribution; and refuelling methods. (UK)

NSA, vol. 25, 1971

14586 DEVICE FOR SUSPENDING FUEL PINS IN AN ASSEMBLY. Agranier, Jean-Claude; Cayol, Andre (to Commissariat a l'Energie Atomique). French Patent 2,063,107. 9 Jul 1971. Filed 19 Sep 1969. (In French).

A device for suspending fuel pins in an assembly is described. The bundle of pins, which are distributed regularly in a lattice, is supported in an envelope in which the coolant flows. Two parallel supporting rods are attached to the envelope; they are positioned transverse to the direction of the flow. Rails slide on these rods and through the heads of the fuel pins. (France)

NSA, vol. 26, 1972

14587 DEVICE FOR SUSPENDING FUEL PINS. Agranier, Jean-Claude; Chalony, Andre (to Commissariat a l'Energie Atomique). French Patent 2,060.226. 18 Jun 1971. Filed 17 Sep 1969. (In French).

A device for suspending a bundle of fuel pins in a fuel assembly is described. The pins are distributed regularly in a lattice and are contained in an envelope in which the coolant flows. Two fixed parallel axes traverse a tube held in this envelope. Rails slide through these rods and extend further than the tube edge. Guide pieces slide on each rail and through the heads of the fuel pins. (France)

NSA, vol. 26, 1972

1. Status Report on Boiling-Water Reactor Fuels*, *R. K. Andersen, J. T. Mommsen (GE-San Jose)*

Extensive operating experience has shown Zircaloy-clad UO_2 to be the most reliable and economical fuel for boiling-water reactors. To date, over 7.9 billion kWh of electricity have been generated from power produced by this type of fuel. The Dresden Type-I fuel has reached peak assembly exposures of 25 000 MWd/t (June 30, 1966), and some of these assemblies have been in service over six years, longer than projected requirements for BWR fuels.

Fuel cycle costs have been reduced to 17 to 18 cents per million Btu for first core loads, declining to 12 to 13 cents in third cores.

Over 170 000 Zircaloy-2-clad fuel segments have been irradiated in boiling-water power reactors. There have been only ten in-service defects, and these occurred in the Dresden first core fuel. The defects were small cladding penetrations caused by minute cracks and localized overheating and corrosion resulting from rod bowing. The plant continued to operate until scheduled refueling; operation never was jeopardized. The causes of the defects were corrected and no subsequent defects have been found in any operating reactors.

The trend in BWR fuel design is to higher core power density (38% increase since Dresden 2) and longer fuel life (19 000 MWd/t first core, 25 000 MWd/t equilibrium core). Burnable poisons will be used to permit the higher exposures without an increase in reactivity worth of control rods. Both high power density and the burnable poison concept have been demonstrated in operating boiling-water power reactors. Erbia and gadolinia are being used as burnable poisons in Dresden, and gadolinia is being irradiated in Big Rock Point. The Big Rock Point plant has achieved a power density of 60 kW/liter.

Lower material and fabrication costs will be a major source of improvement in BWR fuel-cycle costs. One such improvement is the introduction of Zircaloy fuel-rod spacers to enhance neutron economy in the core.

Although standard BWR fuel is fabricated from sintered UO_2 pellets, a promising new process, vibratory compaction, offers potentially lower fabrication costs. Several hundred Zircaloy-2, compacted-powder fuel rods are under irradiation in the Dresden, Big Rock Point, and JPDR reactors. So far, pellet and powder fuel have performed equally well.

Plutonium generated in BWR cores will be recycled for optimum fuel-cycle economy. Physics studies show that direct substitution of plutonium for uranium enrichment is compatible with current boiling-water reactor designs. Development programs for plutonium recycle sponsored by Edison Electric Institute are underway to 1) establish a value for plutonium in BWR recycle applications, 2) develop an economical fabrication technique, and 3) explore alternate methods for use of plutonium in a BWR core. The results of these studies will be verified in test irradiations, followed by large-scale core irradiations in a power reactor.

*Sponsor: Paul Dragoumis

ANS TRANS., vol. 9, 1966

49580 (ANL-7410, pp 47-72) THERMAL REACTOR PHYSICS. (Argonne National Lab., Ill.).

ARGONNE ADVANCED RESEARCH REACTOR — core for, effective neutron multiplication factor calculations for HFIR-type
— operation of, calculations of xenon buildup after power setback from equilibrium
— core for, calculated group-averaged eta values and energy distributions in HFIR-type
— core for, calculated thermal neutron fluxes at midplane in HFIR-type
— thermal column for, calculated effects of core size and loadings on thermal fluxes in internal
COPPER — cadmium ratio measurements in EBWR at 40 and 70 MW
CRITICAL ASSEMBLIES — cores for, revised conversion ratios for 3% and 5% enriched uranium oxide (UO_2) fueled
EXPERIMENTAL BOILING WATER REACTOR — core for, description and loading diagrams for plutonium recycle
— cores for, comparison of activation distributions in plutonium fueled and uranium fueled
— fuel elements for, results of post-shutdown γ scanning of plutonium recycle
GOLD — cadmium ratio measurements in EBWR at 40 and 70 MW
LUTETIUM ISOTOPES Lu-176 — cadmium ratio measurements in EBWR at 40 and 70 MW
LUTETIUM ISOTOPES Lu-177 — cadmium ratio measurements in EBWR at 40 and 70 MW
NEUTRONS — activation ratios for uranium oxide (UO_2) fueled-water moderated assemblies, modified equations for sub-cadmium and thermal
— multiplication factors in HFIR-type cores, calculation for AARR of
— eta values for AARR HFIR-type core, calculations of group averaged
— energy distributions in AARR HFIR-type core, calculations of group-averaged
— fluxes in AARR internal thermal column, calculated effects of core size and loading on
REACTOR FUEL ELEMENTS — gamma scanning of EBWR plutonium recycle, results of post-shutdown
XENON — buildup in AARR, calculations of effects of power setback on (M.L.S.)

NSA, vol. 23, 1969

9082 CONTINUOUS SINTERING FURNACE. Aubry, Bernard; Gillot, Jacques; Masselot, Yves; Teboul, Albert (to Commissariat a l'Energie Atomique). French Patent 2,055,946. 14 May 1971. Filed 8 Aug 1969. (In French).

A furnace and the associated equipment used for sintering pellets of ceramic nuclear fuel in a neutral or reducing atmosphere is described. The process is intended particularly (although not exclusively) for pellets of uranium oxide and/or plutonium. It consists of a tubular enclosure with a horizontal axis, provided with a means of being heated along its length, a hearth for moving longitudinally the boats containing the pellets. A circuit is incorporated for maintaining a neutral or reducing atmosphere and for extracting the bonding agent. This circuit consists of a means of injecting the neutral or reducing gas into the two ends of the furnace, a means of removing the gas from the intermediate region between the pre-sintering and sintering regions, and a means of condensing the bonding agent which is found in this region. The system possesses a means of introducing the boats at one end of the furnace, moving them along the hearth, and extracting them from the other end of the furnace. (France)

NSA, vol. 26, 1972

58671 (CONF-720836-1) NUCLEAR FUEL CYCLE. Brooksbank, R. E. (Oak Ridge National Lab., Tenn.). 1972. 41p. Dep. NTIS.

From Energy source for the future symposium; Oak Ridge, TN. (18 Aug 1972).

Specific plant operating experiences in parts of the cycle and regulatory means now used to control the prime operational areas of the fuel cycle and to minimize their potential insult to the public from a health and safety standpoint are outlined. Uranium ore processing, production of UF_6, enrichment of ^{235}U, conversion to UO_2, and fabrication are discussed. Fuel reprocessing is also considered. Hazards potentials in fuel transport, reprocessing, fuel preparation, and waste handling are discussed. An oxide conversion line used to provide ceramic-grade $^{233}UO_2$ is described. Six years of operational experience in reprocessing by Nuclear Fuel Services at West Valley, New York are reviewed. Manpower requirements for the fuel cycle are considered. Future nuclear fuel cycle problems are postulated. (M.C.G.)

NSA, vol. 26, 1972

1. Criticality Safety Considerations in the Fabrication of UO_2 and PuO_2-UO_2 Fuels for Light-Water Reactors, *C. L. Brown (PNL), Invited*

Commercial fabrication of uranium oxide fuels for light-water reactors is the fastest maturing segment of the nuclear fuel cycle. Some ten commercial fuel fabricators now routinely manufacture uranium fuels on a more or less mass production basis. With this maturing comes an increased incentive to increase production rates and thereby reduce fuel fabrication costs. One astutely observes that the only restraints that prevent unlimited fuel throughput are the limits imposed to ensure criticality safety. It, therefore, behooves us to periodically reexamine plant equipment in light of advances in criticality safety technology and to adjust limits wherever possible to enhance the economics of the fuel cycle.

This paper will discuss some of the criticality safety aspects of uranium oxide fuel fabrication, and present new information on plutonium oxide/uranium oxide fuel fabrication, which is now in the developmental production-line stage.

Uranium oxide fuel fabrication plants all use the same basic techniques and processes. Uranium is received as UF_6; the UF_6 is converted to UO_2, most commonly by an aqueous process; the UO_2 is pressed into pellets; the pellets are loaded into rods; and the rods are combined into fuel assemblies.

The most challenging part of the process from the criticality safety viewpoint is the conversion of UF_6 to UO_2. Most widely used in commercial facilities is the ammonium diuranate (ADU) process.[1,2] The chemistry of this process involves two steps:

1. hydrolysis of UF_6 with ammonium hydroxide: $UF_6 + 4NH_4OH \rightarrow UO_2F_2 + 4NH_4F + 2H_2O$.
2. precipitation of the hydroxide and formation of ADU: $UO_2F_2 + 2NH_4OH \rightarrow UO_2(OH)_2 + 2NH_4F$ $2UO_2(OH)_2 + 2NH_4OH \rightarrow U_2O_7(NH_4)_2 + 3H_2O$.

In the subsequent heating step, the ADU undergoes thermal decomposition with simultaneous reduction to UO_2.

Another production method used for the conversion of the UF_6 to UO_2 is the ammonium uranyl carbonate (AUC) process. The form of the AUC precipitate is probably $(NH_4)_4[UO_2(CO_3)_3]_9$. This compound is calcined to UO_3 or U_3O_8, which is then reduced[3] to UO_2.

There has been mention, but no detail, of a new shorter and more economical dry process for the conversion of UF_6 to UO_2, which has been successfully developed in the UKAEA. A 300 ton/year production line is currently under production based on this process.[4] The fact that this method is dry implies a tremendous production rate advantage over the aqueous processes, because a major part of the criticality safe bottleneck is eliminated.

Uranium fuel fabrication covers the full spectrum of criticality safety technology. UF_6 is handled in ton quantities, safe by virtue of undermoderation; UF_6 to ADU or

ANS TRANS., vol. 14, 1971

AUC conversion equipment is strictly limited because of optimum moderation; fabrication of pellets, rods, and assemblies takes into account heterogeneity, and again, undermoderation; and neutron interaction in all areas affords a challenge to the analyst. Of these problems, the one that will be stressed in this paper is the importance of density in the ADU process.

Plutonium oxide/uranium oxide fuel fabrication for light-water reactors will require the development of new fabrication equipment and techniques. The fabrication steps involved will be: blending of uranium and plutonium, which could be a dry UO_2 - PuO_2 process or an aqueous mixing or precipitation process; pellet pressing; rod loading; and fuel element assembly. The challenge to the development of equipment will be presented primarily by radiation safety, rather than criticality safety. Radiation safety will require that all operations from mixed-oxide blending to fuel rod loading by carried out in sealed glove boxes. Calculated criticality parameters for the slightly enriched homogeneous PuO_2-UO_2 system will be presented. Parameters for the similar mixed-oxide heterogeneous rod system have been reported.[4]

---

1. C. D. HARRINGTON and A. E. RUEHLE, *Uranium Production Technology*, Mallinckrodt Chemical Works, D. Van Nostrand Company, Inc., Princeton, N.J. (1959).
2. R. VUILLERNEY, "Contribution to the Chemical and Technological Study of Ammonium Diuranate Precipitation," CEA 2204, Centre D'Etudes Nucléaires De Saclay, France (1962).
3. M. JENSEN, "Investigations on Methods of Producing Crystalline Ammonium Uranyl Carbonate and Its Applicability for Production of UO_2 Powder and Pellets," RISO 153, Danish AEC, Risoc Research Establishment (1967).
4. C. L. BROWN and D. R. ODEN, "Critical Parameters for 0 to 7% PuO_2-U(nat) O_2 Rods in Water," *Trans. Am. Nucl. Soc.*, 12, 873 (1969).

ANS TRANS., vol. 14, 1971

10251 PLUTONIUM–URANIUM DIOXIDE POWDER AND PELLET FUEL MANUFACTURE. Caldwell, C. S.; Puechl, K. H. (Nuclear Materials and Equipment Corp., Apollo, Pa.). Nucl. Met., Met. Soc. AIME, 13: 174-94(1968).

From the Nuclear Metallurgy Conference, Phoenix, Ariz. See CONF-671005.

Plutonium–uranium dioxide powder preparation and pellet fuel manufacturing methods have been developed to meet increasing demands for critical assembly, test reactor, and demonstration power reactor fuel elements. A review of current conversion and fabrication processes is presented and techniques for control of mixed-oxide pellet characteristics including plutonium microhomogeneity and oxygen/metal ratio are discussed. Experience gained during fabrication of thermal recycle and fast breeder type PuO_2–UO_2 fuel materials in quantities up to 500 kg is presented. Product variability data are included. (auth)

NSA, vol. 23, 1969

11359 FUEL SUBASSEMBLY FOR A NUCLEAR REACTOR. Chubb, Walston; Keller, Donald L.; Wullaert, Richard A.; Storhok, Victor W. (to United States Atomic Energy Commission). U. S. Patent 3,619,366. 9 Nov 1971. Filed 6 May 1969.

A fuel subassembly for a nuclear reactor incorporating a body of uranium dioxide or mixed uranium–plutonium dioxide fuel within an elongated casing is described. A gas plenum is located at the top and bottom of the subassembly and a metal tube or rod serving as a heat sink extends along the axis of the subassembly. (Official Gazette)

NSA, vol. 26, 1972

2601 PROCESS FOR PREPARING SINTERED PELLETS OF CERAMIC MATERIALS TO BE USED AS NUCLEAR FUELS AND PRODUCTS OBTAINED BY THE PROCESS. (to Comitato Nazionale per L'Energia Nucleare). British Patent 1,241,976. 11 Aug 1971. Priority date 4 Aug 1967, Italy.

A process is described for preparing sintered pellets of ceramic materials, such as U, Th, and Pu oxides, singly or in admixture with one another to be used as nuclear fuels. The sintered pellets have a specific surface area not greater than 1 m^2/g and a particle size smaller than 40 μ. The pellets are pressed without any preliminary mixing with binders and lubricants and then sintered in a non-oxidizing atmosphere free from hydrocarbons at a temperature in the range from 1150 to 1600°C for a period of from 1 to 8 hr. (P.C.H.)

NSA, vol. 26, 1972

31391 (ARH-470) LABORATORY STUDIES OF MIXED-OXIDE POWDER PREPARATION BY CONTINUOUS AMMONIA PRECIPITATION. Curtis, M. H. (Atlantic Richfield Hanford Co., Richland, Wash.). 1 Apr 1968. Contract AT(45-1)-2130. 38p. Dep. NTIS.

Studies are continuing on the continuous coprecipitation of uranium diuranate–plutonium hydroxide by ammonia of a product that would meet powder quality specifications. The goal was twofold: to achieve a high filtration rate for the slurry (relative to plutonium oxalate, and in the same type of continuous precipitator) on a vacuum drum filter; and to achieve satisfactory hydrogen reduction in a continuous vibrating tube reactor so as to give acceptable powder properties. Experimental work in laboratory and prototype equipment established that the existing plutonium finishing equipment has the potential of producing mixed oxides by the ammonia coprecipitation process. Continuous precipitation of a slurry suitable as feed for a continuous drum filter is possible by using close pH control and elevated temperature. Continuous hydrogen reduction may be achieved in a vibrating tube reactor if the equipment is capable of providing suitable temperatures, gas flow rates, and residence times. By following the techniques developed, the product oxide physical and chemical properties are suitable for fuel fabrication. (M.C.G.)

NSA, vol. 26, 1972

3,652,744

METHOD OF MAKING NUCLEAR FUEL ELEMENTS

Richard C. Dahlberg, La Jolla, and Walter V. Goeddel, San Diego, Calif., assignors to the United States of America as represented by the United States Atomic Energy Commission

No Drawing. Filed Nov. 19, 1969, Ser. No. 878,198
Int. Cl. G21c *21/02*
U.S. Cl. 264—.5 **3 Claims**

Nuclear fuel elements which contain different amounts of nuclear fuel in the form of minute particulate nuclear fuel cores having fission product-retentive coatings are made using casings which have the same amount of fuel chamber volume for holding the nuclear fuel. First and second groups of coated fuel particles are used which vary from each other either in core composition, core size, or coating thickness and thereby contain different amounts of nuclear fuel per unit volume of coated particles. Thus, different fuel elements contain different amounts of nuclear fuel as a result of containing different relative amounts of coated nuclear fuel particles from the first and second groups.

U.S. PATENTS 1972

19428 FESSENHEIM FUEL DESIGN AND MANUFACTURE. Darolles, J. F.; Leclercq, J. (Societe Franco-Americaine de Constructions Atomiques, Courbevoie, France). Nucl. Eng. Int.; 20: No. 234, 675-678(Sep 1975).

The fuel elements in first cores of Fessenheim 1 and 2 consisted of 15 × 15 pin arrays. After revision of the ECCS criteria in 1973 a new 17 × 17 pin array was favored. Implementation of this concept involved design tasks and adaptation of manufacturing processes and procedures. Specific qualification programs which were developed for the manufacturers are described. These concerned the production of uranium dioxide pellets and zircaloy-4 cladding tubes. An integrated system, set up to administer and analyze the quality of the manufactured fuel included a traceability system for locating and identifying the fuels, and quality evaluation during manufacturing. (UK)

NSA, vol. 33, 1976

48339 PRODUCTION-SCALE FORMATION OF ULTRAHIGH PURITY CARBIDE POWDERS. Davidson, Keith Vernon; Schell, Donald H. (to United States Atomic Energy Commission). U. S. Patent 3,666,845. 30 May 1972. Filed 6 May 1970.

A method is described for forming ultrahigh purity carbide powders (UC–ZrC) on a production scale, which comprises blending metal oxide powders with elemental carbon powders and a partially polymerized furfuryl alcohol binder together with a suitable polymerization catalyst, extruding, polymerizing the binder, heating to 840°C for 48 hours in a soft vacuum, heating to 2,200°C for 1 hour in a helium atmosphere, cooling, crushing, and then heating to 2,000°C for 24 hours under a vacuum. (Official Gazette)

NSA, vol. 26, 1972

15169 METHOD OF PREPARATION OF NUCLEAR FUEL BODIES. Davis, D. E.; Stevens, D. W.; Tully, G. R. Jr. (to Gulf Oil Corp.). German Patent 1,906,269. 6 Apr 1972. (In German).

A method is described for preparing a solid nuclear fuel body where a chamber having the approximate size and shape of the desired nuclear fuel (e.g., a cavity in a nuclear fuel element part of porous graphite) is used. First, the chamber is fed with the nuclear fuel as loose single particles according to the desired density and then the binder (e.g., heat setting resin) in liquid form under pressure is forced into the chamber in such a manner that the binder flows through the filled particles of nuclear fuel and fills the chamber uniformly, this occupying the vacancies between the fuel particles. The uniform aggregate of fuel particles and binder is then heated in the chamber to harden the binder. Finally, the aggregate undergoes pyrolysis outside the chamber. (GF)

NSA, vol. 27, 1973

9855 METHOD OF MANUFACTURING SINTERED SUBSTOICHIOMETRIC OXIDES. Dean, Guy (to Commissariat a l'Energie Atomique). French Patent 2,061,557. 25 Jun 1971. Filed 27 Jun 1969. (In French).

A method of manufacturing ceramic fuels, consisting of mixed substoichiometric dioxides of U and Pu, is described. A mixture (in powdered form) of the oxides and carbon is sintered, the appropriate proportions being used to obtain the required substoichiometry on eliminating oxygen in the form of CO. The sintered material is then crushed, before undergoing a second sintering in conditions identical to those of the first sintering. The sintering takes place between 1500 and 1800°C in a vacuum of at least 10^{-4} torr. (France)

NSA, vol. 26, 1972

CEA Light-Water Fuel Experiments in the BR3, *Ghislain de Contenson, Guy Lestiboudois, Michel Watteau (CEA–France)*

INTRODUCTION

With a view to qualifying the light-water fuels manufactured by the CEA, 11 assemblies of pins were or are being irradiated in the BR 3 reactor at Mol in Belgium. All the pins performed well and no burst claddings occurred in these assemblies.

The CEA experiments in the BR 3 took place in three successive stages:

1. Four assemblies were irradiated in the BR 3-2bis core from July 31, 1969 to December 21, 1970.
2. Two assemblies are under irradiation in cores BR 3 3-A and BR 3 3-B (since September 24, 1972).
3. Five assemblies are under irradiation in the BR 3 3-B core (since July 2, 1974).

At the present time, only the first four assemblies have been unloaded and examined in the hot laboratory.

IRRADIATION OF FOUR ASSEMBLIES IN CORE 2 bis

Each assembly comprised 36 fuel pins with the following characteristics:

UO_2	outside diameter:	7.42 mm
	density:	10.33
	stacked length:	≃ 1000 mm
	pellets ground and as-sintered	
Cladding	Zircaloy 4	
	outside diameter:	8.65 mm
	thickness:	0.45 mm

Length of pins: 1250 mm
internal filling: 1 bar of helium
plug-cladding welded by electron beam welding or argon-arc.

After fabrication, the pins were externally oxidized in an autoclave.

The irradiation took place from July 31, 1969 to December 21, 1970. The power of the reactor had to be limited (owing to cladding failures in assemblies from the previous core), but despite this, the operating characteristics are very interesting:

Mean burnup of the assembly: 18,500 MWd/MTU
Mean burnup of the most charged pins: 23,400 MWd/MTU (with a maximum 26,700 MWd/MTU)
Integrated fast flux $E > 1$ MeV: 1.6×10^{21} n/cm^2
Maximal linear power: 440 W/cm
Fluid pressure: 140 bars

ANS TRANS., vol. 20, 1975

Hot laboratory examinations were carried out on the four assemblies and revealed the following points:

1. Little adherent deposits. After cleaning, the claddings had a good aspect similar to that before irradiation.
2. Elongation of pins between 0.1 and 0.2%.
3. Systematic shortening of the length of the oxide columns through densification, around 1%.
4. The Zy-4 claddings showed diametral creep deformation of 0.08%.
5. Very slight cladding hydridation with a 90-ppm maximum hydrogen content.
6. The maximum fraction of the fission gas quantities released is 11%.
7. Identical behavior of as-sintered and ground UO_2 pellets.
8. Temperatures in the middle of the UO_2 did not exceed 1400°C.
9. A very localized UO_2-cladding reaction zone appears on the pins with a burnup exceeding 22,000 MWd/MTU.

These examinations confirm the excellent behavior of these pins whose geometry and service conditions are very like those of the fuel elements envisaged at the present time in pressure water reactors being built in France.

IRRADIATION OF TWO ASSEMBLIES IN THE 3A and 3B CORES

Each assembly was fitted with 20 pins and 3 grids.

Characteristics of the Pins

There are three different types:

Type I: Arc welded pins, fitted with solid laterally ground UO_2 pellets, sealed under 10 bars absolute of helium.

Type II: Resistance welded pins, fitted with solid laterally ground UO_2 pellets, sealed under 10 bars absolute of helium.

Type III: Resistance welded pins, with as-sintered annular pellets, sealed under 2 bars absolute of helium.

UO_2: outside diameter = 9.19 mm
inside diameter = 2.5 mm
density = 10.30
stacked length = 985 mm

Cladding: Zircaloy 4
Two cladding batches (1 batch equipping Types I and II) (1 batch equipping Type III)

Nominal dimensions:
inside diameter = 9.48 mm
thickness = 0.65 mm

IRRADIATION OF THE FIVE ASSEMBLIES IN CORES III-B and IV

The main characteristics of these assemblies are given below:

1. twenty-eight UO_2-Zircaloy pins, assembled, 17 × 17 geometry cladding (outside diameter: 9.40 mm; thickness: 0.6 mm) "stable" UO_2 pellets

2. pins distributed in a square lattice with a 12.6-mm pitch (identical to the 17 × 17 arrangement)

3. four assemblies of nonpressurized pins (sealed under one bar absolute of helium) and 1 assembly fitted with pressurized pins (sealed under 35 bars absolute of helium), to evaluate the effect of pressurization on the behavior of the pins.

Three different types of cladding, expected to give better behavior under creep compared with the claddings used so far, were tried.

Grid Characteristics

These are in Inconel 718. Their mounting is original in design allowing them to follow the movements of the fuel pin bundle, whether due to heat expansion or elongation during irradiation.

The irradiation of these two assemblies took place in two stages:

1. *Core 3 A from September 24, 1972 to January 11, 1974*: The power of the reactor was reduced December 27, 1973 due to excessive activity being detected in the primary system as a result of cladding failures. The "dry sipping" effected on our assemblies showed that all the rods were leaktight.

The following burnups were reached:

	No. 1	No. 2
Average assembly burnup, MWd/MTU	17,800	21,700
Maximum pin burnup, MWd/MTU	21,000	23,000
Maximum pellet burnup, MWd/MTU	27,000	29,900

The maximum linear power was 485 W/cm.

2. *Core 3 B—Now Proceeding*: These two assemblies were recharged in core 3 B operating since July 2, 1974. On termination of the III-B irradiation, these two assemblies will reach average burnups of 21,000 and 30,000 MWd/MTU, and one of the two will probably be sent to the hot laboratory for destructive testing, whereas the other will continue in core IV-A.

The linear power aimed at is around 450 W/cm, the burnup, during cycle 3-B, could approach 10,000 MWd/MTU.

Irradiation started on July 2, 1974 and by July 23 the reactor reached 80% of its power. At the present time, the reactor is at service power and after one month in operation, no anomaly has been noticed.

CONCLUSION

All irradiations in the BR 3 reactor were very instructive for CEA light-water fuels, for they enabled new concepts and new methods of fuel pin fabrication to be qualified. The results of the irradiation of the four 17 × 17 geometry assemblies in the BR 3-2bis core as from 1970, at a power very close to that now aimed for in electronuclear reactors, are very important for understanding the behavior of pins with this geometry.

ANS TRANS., vol. 20, 1975

20745 HOT PRESSING OF UO_2–PuO_2. de Freitas, C. T.; Ayer, J. E.; Petkus, E. J. (Argonne National Lab., Ill.). Metalurgia (Sao Paulo); 26: No. 148, 183-7(Mar 1970). (In Portuguese). (IEA-225).

Studies of 85% UO_2–15% PuO_2 hot pressing in the 1000 to 1500°C temperature range are described. The operations were conducted in graphite dies at variable pressures between 110 and 290 kg/cm^2. The temperature influence on volume decrease and density was determined. For the oxide mixture considered, the largest volume decrease rate was found at 930°C. 19 mm side square plates and 25.4 mm diameter discs were produced. Three plates were prepared by multispecimen hot pressing. In spite of the graphite die utilization, the carbon absorption in the not pressed plates was generally under 500 ppM. The ceramographic study showed that the carbon was distributed principally near the samples surface. The oxygen/metal ratio decreased during the pressing operations. The largest density values, corresponding to 95% of the theoretical value, were obtained for an operation conducted at 1500°C for 30 minutes, at 270 kg/cm^2 pressure. Ceramography showed that good UO_2–PuO_2 mixture homogeneity can be obtained. (auth)

NSA, vol. 26, 1972

22792 SINTERING URANIUM OXIDE IN THE REACTION PRODUCT OF HYDROGEN-CARBON DIOXIDE MIXTURES. De Hollander, W. R.; Nivas, Y. (to General Electric Co.). US Patent 3,872,022. 18 Mar 1975. Filed date 9 May 1973. 10p.

Compacted pellets of uranium oxide alone or containing one or more additives such as plutonium dioxide, gadolinium oxide, titanium dioxide, silica, and alumina are heated to 900 to 1599°C in the presence of a mixture of hydrogen and carbon dioxide, either alone or with an inert carrier gas and held at the desired temperature in this atmosphere to sinter the pellets. The sintered pellets are then cooled in an atmosphere having an oxygen partial pressure of 10^{-4} to 10^{-18} atm of oxygen such as dry hydrogen, wet hydrogen, dry carbon monoxide, wet carbon monoxide, inert gases such as nitrogen, argon, helium, and neon and mixtures of any of the foregoing including a mixture of hydrogen and carbon dioxide. The ratio of hydrogen to carbon dioxide in the gas mixture fed to the furnace is controlled to give a ratio of oxygen to uranium atoms in the sintered particles within the range of 1.98 : 1 to about 2.10 : 1. The water vapor present in the reaction products in the furnace atmosphere acts as a hydrolysis agent to aid removal of fluoride should such impurity be present in the uranium oxide. (Official Gazette)

NSA, vol. 32, 1975

38877 REACTION FUEL (OXIDE MIXTURE $(UPu)O_{2\pm x}$) CLADDING INTERFACE (STAINLESS STEEL) IN IRRADIATED FUEL ELEMENTS. de Keroulas, F.; le Beuze, R.; Calais, D.; van Craeynest, A.; Conte, M. (CEN, Fontenay-aux-Roses, France). J. Nucl. Mater.; 43: No. 3, 313-20(Jun 1972). (In French).

During operation of a reactor, fission causes oxidation of the fuel $(U,Pu)O_{2-x}$. For a certain value of x, which is a function of the composition of the steel of the cladding, thermodynamic considerations imply oxidation of the cladding by the fuel. This reaction is limited by the very low mobility of chromium and oxygen in oxides of the ferrite type, $(Cr,Fe)_2O_3$, which form at the fuel/cladding interface. Nevertheless the reaction may proceed by the intermediary of volatile fission products, the more so where the internal stresses due to swelling of the fuel and to the thermal gradient may cause cracking of the protective oxide layer. Tellurium reacts with the steel to yield a phase rich in chromium and manganese (19 and 6 wt% respectively). A superficial reduction in chromium content results and this reduces the oxidation resistance of the steel. Iodine leads to vapor-phase transport. Cesium iodide (a compound of fission products) attacks the grain boundaries in the steel by depositing cesium while forming a mixture of iron iodide and chromium iodide vapors. These volatile iodides migrate towards hotter zones and decompose in regions at temperatures around 900°C. (auth)

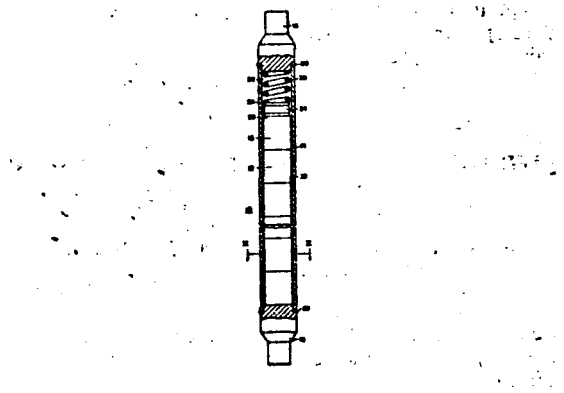
NSA, vol. 26, 1972

38847 PROCESS FOR THE PREPARATION OF SPHERICAL PARTICLES OF NUCLEAR FUEL. (to EURATOM). British Patent 1,273,948. 10 May 1972. Priority date 10 Apr 1970, Netherlands.

A sol-gel process for preparation of spherical particles of nuclear fuel consisting of mixed oxides and/or carbides and/or nitrides of U and Pu comprises mixing a concentrated denitrated solution of uranyl nitrate with a Pu hydroxide solution and an ammonia donor and gelling droplets of the mixture to form the spherical particles. Finely divided C must also be present when carbides and nitrides are required in the spherical particles. The nitrides can be formed by heating particles obtained from a C-containing mixture in a N_2 atmosphere. (B.L.M.)

NSA, vol. 26, 1972

3,677,894
INTERNALLY PRESSURIZED FUEL ELEMENT
Harry M. Ferrari, Pittsburgh, Pa., assignor to Westinghouse Electric Corporation, Pittsburgh, Pa.
Continuation-in-part of application Ser. No. 706,291, Feb. 19, 1968, which is a continuation-in-part of application Ser. No. 615,542, Feb. 13, 1967. This application Aug. 14, 1969, Ser. No. 850,198
Int. Cl. G21c *3/18*
U.S. Cl. 176—68 10 Claims



A fuel element for a pressurized nuclear reactor comprising a sealed cladding case, nuclear fuel therein and means for producing an internally pressurized atmosphere so that the fuel element is free standing, and characterized by metal wall cladding having a reduced thickness for conditions of reactor use.

U.S. PATENTS 1972

23228 STABILIZED ZIRCONIUM DIOXIDE AND HAFNIUM DIOXIDE COMPOSITIONS. Fehrenbacher, Larry L. U. S. Patent 3,634,113. 11 Jan 1972. Filed 30 Oct 1968.

A Type C mixed rare earth oxide solid solution is used to eliminate the monoclinic phase of zirconium dioxide and thus produce a stable refractory. The Type C solid solution consists primarily of oxides of dysprosium, erbium, ytterbium, and holmium with small amounts of thulium, terbium, and lutetium. This solid solution is applicable to hafnium dioxide and to mixtures of hafnium dioxide with zirconium dioxide as well. (Official Gazette)

NSA, vol. 26, 1972

15672 (KFK-1418) SPECIAL RESEARCH METHODS IN THE FIELD OF REACTOR MATERIAL. Short Collection of Seminar Proceedings, Winter 1970/71. Frieser, H. (comp.) (Kernforschungszentrum, Karlsruhe (West Germany). Institut fuer Material- und Festkoerperforschung). Jun 1971. 30p. (In several languages). Dep. NTIS (U. S. Sales Only).

Eight seminar papers on research methods are compiled. They include analysis of plutonium-containing and irradiated fuels, investigation methods in the hot cells of reactor plants, investigations on the accuracy of electron beam microanalysis, quantitative microscopic texture analysis, nucleation phenomena in gas-metal reactions, investigation methods for plutonium-containing fuels, review of analysis methods used in chemical analytical laboratories with emphasis on modern methods, and particle size measurement by electron microscopy in the size range 500 Å to 2μ. (J.S.R.)

NSA, vol. 26, 1972

9846 IMPROVEMENTS IN THE PREPARATION OF SINTERED URANIUM DIOXIDE GRAINS. (to General Electric Co.). British Patent 1,256,283. 8 Dec 1971. Priority date 29 Oct 1968, United States.

A method is described for the production of sintered UO_2 grains of controlled size for use as nuclear fuel. Uranium dioxide or ammonium diuranate powder is pressed into compacts and then sintered at a temperature of at least 1900°C until grains of UO_2 of the desired size are formed. The sintering is carried out in an atmosphere of dry hydrogen with a dew-point sufficient to produce sub-stoichiometric UO_2. The sintered compacts are then cooled in an atmosphere of hydrogen or an inert gas to precipitate metallic uranium at the boundaries of the grains, and when the temperature reaches about 400°C, the cooling is continued in a hydrogen atmosphere such that the precipitated metallic uranium reacts with the hydrogen to form uranium hydride, thereby causing the sintered compacts to disintegrate into substantially the grains of UO_2. The product is free-flowing and can be combined readily with other materials if desired. (P.C.H.)

NSA, vol. 26, 1972

6274 (NEDC-12198) PLUTONIUM UTILIZATION IN BOILING WATER REACTORS, PHASE II. Semiannual Report, July-December 1970. (General Electric Co., San Jose, Calif. Reactor Fuels and Reprocessing Dept.). May 1971. 39p. For Edison Electric Inst., New York.

The development program for utilization of plutonium in boiling water reactors for the period July to December 1970 is reviewed. Big Rock Point reactor irradiation performance results are reported for 28 PuO_2-UO_2 fuel rods and three all PuO_2-UO_2 fuel rod bundles. Hot cell examination results to date are reported for four rods removed from Big Rock Point in February 1970 after one cycle of irradiation. Vermont Yankee design finalization and fuel fabrication for four recycle demonstration fuel bundles are reported and plans for pre-irradiation characterization and bundle assembly are described. Activities associated with the AEC-sponsored Integrated Safeguards Experiment and Plant Instrumentation Program are reported. The plutonium recycle design optimization and economic studies are summarized. The significant conclusions or observations of the program to date are that the test irradiations are proceeding without indication of failure problems; the interchangable plutonium recycle bundle design objective has been accomplished without compromise of important performance, economic, and safeguards criteria; and the annular fuel pellet concept continues to show promise as the most economical design for plutonium recycle in boiling water reactors. (auth)

NSA, vol. 26, 1972

3,880,769

METHOD OF MAKING MICROSPHEROIDAL NUCLEAR FUELS HAVING CLOSED POROSITY

John M. Googin, and Charles R. Schmitt, both of Oak Ridge, Tenn., assignors to The United States of America as represented by the United States Atomic Energy Commission, Washington, D.C.

Continuation of Ser. No. 48,579, May 25, 1970, abandoned. This application Dec. 19, 1972, Ser. No. 316,625

Int. Cl. C09k *3/00*

U.S. Cl. 252—301.1 R 9 Claims

1. A process for forming a nuclear fuel which comprises contacting ion exchange resin beads with an aqueous solution of an inorganic salt of at least one actinide metal, separating the actinide-loaded resin beads from the solution, drying the separated beads, and then carbonizing the dried beads at a temperature in the range 1,400°–2,200°C. to form spheroidal particles with closed porosity containing the actinide metal as an oxide or carbide dispersed within a continuous carbon matrix.

U.S. PATENTS 1975

3,887,486

POROUS, MICROSPHEROIDAL, NUCLEAR FUELS HAVING INTERNAL POROSITY

John M. Googin, and Charles R. Schmitt, both of Oak Ridge, Tenn., assignors to The United States of America as represented by the United States Energy Research and Development Administration, Washington, D.C.

Division of Ser. No. 48,579, May 25, 1970, abandoned. This application Oct. 16, 1972, Ser. No. 298,134

Int. Cl. G21c *3/02*

U.S. Cl. 252—301.1 R 13 Claims

1. A microspheroidal particle consisting essentially of a porous matrix of carbon or graphite, a substantial portion of the porosity of said matrix being of the closed-pore type, and an oxide or carbide of a metal selected from the group uranium, plutonium, and thorium within the internal volume and closed pores of said matrix.

6. Microspheres of uranium carbide produced by loading an ion exchange resin containing ion exchange sites for sorbing uranyl ions, drying the loaded resin and then heating said dried resin in an inert atmosphere at a temperature in the range 1,400°–2,200°C. to convert the resin to a porous carbon matrix containing closed pores and a homogeneously dispersed uranium carbide phase.

12. A microspheroidal particle 5 to 2,000 microns in diameter consisting essentially of a porous carbon matrix and a compound selected from the group consisting of an oxide and a carbide of at least one metal selected from the group consisting of uranium, plutonium, and thorium distributed within said matrix.

U.S. PATENTS 1975

1. Development of a Sol-Gel Process for the Preparation of Dense-Oxide Forms of PuO_2*/

R. G. Haire, M. H. Lloyd (ORNL)

A sol-gel process for the preparation of dense-oxide forms of PuO_2 has been developed and tested on a laboratory scale. This process utilizes the polymerization behavior of tetravalent plutonium to produce crystallites of colloidal size and to maintain valence stability. The final sols are essentially colloidal plutonium dioxide which vary from 1 to 3 M in plutonium and have NO_3^-/Pu mole ratios of 0.1 to 0.3. They are stable for many months and may be stored until needed. These plutonia sols are compatible with thoria sols and urania sols which have been produced at Oak Ridge National Laboratory. Although continued laboratory development work is necessary before engineering scaleup is feasible, the ability to produce dense plutonia as well as homogeneous plutonia-urania or plutonia-thoria bodies at any desired ratio has been demonstrated.

In this process, plutonium is precipitated from dilute plutonium-nitrate solution which contains 1 to 2 M HNO_3 by slow addition of excess NH_4OH with rapid agitation. Essentially complete removal of contaminant ions is readily accomplished by water washing, and the washed precipitate is then peptized by digestion at 50°C with dilute HNO_3. The minimum amount of HNO_3 found to be necessary for complete peptization is one mole of nitrate per mole of plutonium. Although the plutonium sol produced during digestion is a stable colloidal dispersion of tetravalent plutonium polymer, the nitrate-to-plutonium ratio is undesirably high for most fuel cycle applications, and it is reduced to ratios of 0.1 to 0.3 by evaporating the sol to dryness and baking the dried gel at 200°C for about 6 h. The final sol is then prepared by redispersing the baked gel in excess water and evaporating to the desired sol concentration.

Following this procedure, numerous small batches of plutonia sol have been produced with conventional laboratory equipment in batch sizes of 5 to 30 g, Plutonia sols have been dried and calcined to yield dense hard fragments and they have been formed into gel microspheres which calcine to dense-oxide spheres at 1150°C. Plutonia sols have also been mixed in various ratios with thoria and urania sols, and the resultant mixed sols formed into microspheres.

Uniformly sized microspheres are produced in equipment that uses a tapered column. The sols are dispersed as droplets in an organic liquid which converts the droplets to gel spheres by extraction of water. The gel spheres are separated from the organic liquid, dried, and calcined to give the oxide product. The densities of all products obtained to date have varied from 95 to 99% of theoretical, and the crushing strength of the plutonia spheres is in excess of 600 g for a sphere diameter of 150 μ.

---

*Sponsor: R. G. Wymer

ANS TRANS., vol. 9, 1966

12737 REDUCTION/SINTERING OF ADU. Haroun, N. A. Aly, M. M. (Atomic Energy Commission, Cairo). J. Nucl. Mater.; 42: No. 2, 232(Feb 1972).

A preliminary investigation of a process for the reduction/sintering of ammonium diuranate (ADU) pellets into uranium dioxide pellets is described. The process is aimed at eliminating the UO_2 powder preparation step. Pellets of UO_2 were prepared by compressing ADU powders at 10,000 to 70,000 psi and heating in hydrogen at variable rates to the sintering temperature of 1540°C. All prepared UO_2 pellets were either cracked or broken except those compacted at 10,000 psi and receiving a low heating rate. The low heating rate allowed the gases formed during the reduction stage to escape before sintering occurred. (T.F.D.)

NSA, vol. 26, 1972

53546 (AREAEE-134) PREPARATION OF SINTERABLE URANIUM DIOXIDE. Haroun, N. A.; Aly, M. M. E.; El-Adham, K. A. (Atomic Energy Establishment, Cairo (Egypt). Dept. of Metallurgy). 1972. 28p. Dep. NTIS (U. S. Sales Only).

Factors affecting the preparation of sinterable uranium dioxide powders have been extensively and critically reviewed. Experimental work on the reduction of ammonium polyuranate (APU) to UO_2, stabilization by cooling in different media, storage behavior, characterization, and sinterability of the resulting oxides is described. Results indicated that the more stable the powders the higher the reduction temperature. However the deterioration of powder activity and sinterability for the higher temperature preparations led to the choice of an optimum temperature of reduction. Of the cooling media investigated: ethyl alcohol-laden nitrogen (and nitrogen) were more effective than argon and hydrogen in stabilizing the powders. Differential thermal analysis was found most suitable for characterizing the powders. (69 references). (auth)

NSA, vol. 26, 1972

13636 DEVELOPMENT OF STABLE DENSITY UO_2 FUEL. Heal, T. J.; Littlechild, J. E.; Watson, R. H. (BNFL, Springfields, Eng.). pp 52.1-52.3 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004.

Fuel development work in BNFL has produced a technique by which irradiation stable UO_2 of any desired density can be fabricated and the criteria upon which this is based are discussed. (auth)

NSA, vol. 32, 1975

3,907,123

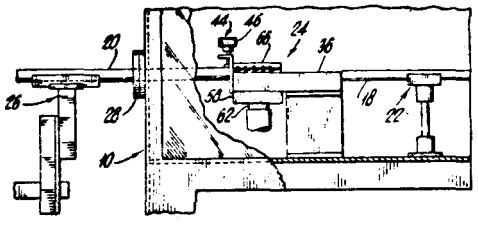
FUEL ROD PELLET LOADING HEAD

Thomas E. Howell, Pasco, Wash., assignor to Exxon Nuclear Company Inc., Bellevue, Wash.

Filed May 31, 1973, Ser. No. 365,507

Int. Cl. B65b *69/00*

U.S. Cl. 214—1 BB **9 Claims**



1. In an assembly for the loading of nuclear fuel pellets into a fuel rod, including a main housing having an inert atmosphere and adapted to receive fuel pellets for loading into elongated fuel rods and a longitudinal extending trough for receiving pellets in end-to-end relationship, wherein the improvement comprises a pellet loading assembly disposed for longitudinal alignment with said trough and said fuel rod, said assembly including a member having a seat for slidably receiving fuel pellets longitudinally from said trough, means for providing relative adjustable vertical alignment between said fuel rod and said seat, said seat being constructed and arranged so that said seat will substantially tangentially contact said pellets for the length thereof only at two locations, whereby the pellets will be transferred from said loading trough to said fuel rod without chipping thereof.

U.S. PATENTS 1975

24873 (GU-5308) MIXED-OXIDE FUEL PELLET FABRICATION AND FUEL ROD ASSEMBLY FOR DRESDEN 1, BATCH 7 RELOAD. Hower, L. D. Jr. (Gulf United Nuclear Fuels Corp., Elmsford, N.Y. (USA)). Jan 1974. 39p. Dep. NTIS $4.00.

The fuel composition is UO_2-2.3 wt% PuO_2. A total of 99 Zircaloy-2-clad fuel rods were produced containing approximately 20,000 pellets. Process data and information are included on fuel pellet fabrication, quality control, fuel rod assembly, fuel powder chemical and physical properties, and Pu recycle pellets. (JRD)

NSA, vol. 29, 1974

25973 ASEA-ATOM BWR: FUEL DESIGN, FABRICATION, AND PERFORMANCE. Hznnera, K.; Hetzler, F.; Hyden, L. Technical Meeting No. 5/6. Basel; Nuclex (1972). 9p.

From international nuclear industries fair; Basel, Switzerland (16 Oct 1972).

Some features of ASEA-ATOM's BWR fuel design and fabrication processes are given. The in pile fuel performance experienoe to date is reviewed. (auth)

NSA, vol. 24, 1974

36497 FABRICATION AND IRRADIATION TESTING OF PuO_2-UO_2 FUELS. Iwasaki, K. (Power Reactor and Nuclear Fuel Development Corp., Tokai, Japan). Genshiryoku Kogyo; 17: No. 12, 30-3(Dec 1971). (In Japanese).

At the Power Reactor and Nuclear Fuel Development Corporation, development work has been in progress on PuO_2-UO_2 mixed fuel, including process and irradiation testing. On production of Pu fuel, processes for fuel pellet production such as coprecipitation and mechanical mixing, and fuel assemblies, as well as plutonium recovery, are described. For test irradiation of the Pu fuel done mainly with facilities in oversea countries, the following tests including those of radiographic, metallurgical, puncture, etc. are explained: fuel produced by the mechanical mixing, the co-precipitation and sol-gel vibration packing; preliminary test for pellets and the sol-gel packed fuel before irradiation in the Halden reactor; and irradiation of fuel assemblies in the Halden reactor. (Japan)

NSA, vol. 26, 1972

24618 FUEL ASSEMBLY FOR A NUCLEAR REACTOR. Jabsen, F. S. (to Babcock and Wilcox Co.). US Patent 3,862,884. 28 Jan 1975. Filed Date 21 Nov 1972. 4p.

A description is given of a nuclear reactor fuel assembly containing a plurality of parallel and adjacent fuel rods and guide tubes. The upper grid of the assembly has parallel rows of T-shaped slots with the upper end of each fuel rod having a T-shaped key for supporting engagement within the T-shaped slots. (Official Gazette)

NSA, vol. 31, 1975

3,897,673

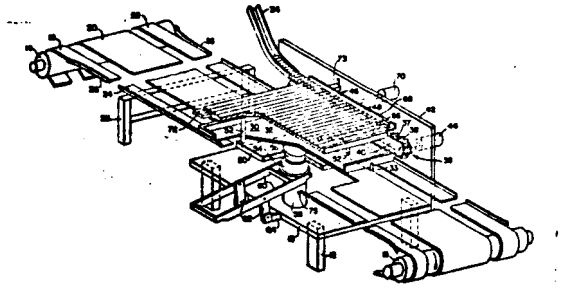
NUCLEAR FUEL PELLET LOADING MACHINE

Robert W. Kee, Gibsonia, and John V. Denero, Leechburg, both of Pa., assignors to Westinghouse Electric Corporation, Pittsburgh, Pa.

Filed Aug. 3, 1973, Ser. No. 385,527

Int. Cl. B65b 57/10

U.S. Cl. 53—61 **9 Claims**



1. Nuclear fuel pellet loading apparatus comprising:

a base supporting a removable carrier adapted to hold nuclear fuel pellets;

means cooperating with said base for moving said carrier into a predetermined position on the base;

stop means on said base engageable by said carrier for stopping movement of the carrier at said predetermined position;

pellet feed means adjacent said base for conveying pellets from a source to a position alongside said carrier; and

pellet push means alongside said base for transferring the pellets from the feed means onto said carrier, the arrangement being such that upon loading the carrier with pellets the means cooperating with the base moves the carrier away from said predetermined position.

U.S. PATENTS 1975

8561 (BMI–1918) PROGRESS ON DEVELOPMENT OF FUELS AND TECHNOLOGY FOR ADVANCED REACTORS DURING JULY 1970 THROUGH JUNE 1971. Annual Report, Task 6 (Part 1) and Task 7. Final Report, Task 6 (Part 2). Keller, Donald L. (Battelle Columbus Labs., Ohio). Jul 1971. Contract W-7405-eng-92. 324p. Dep. NTIS.

EXPERIMENTAL BREEDER REACTOR-II—fuel pins for, fabrication and irradiation testing of plutonium nitride–uranium nitride

LIQUID METAL FAST BREEDER REACTOR—fuels for, development of plutonium nitride–uranium nitride, (E)

—fuels for, high temperature irradiation performance of, (E/T)

—fuels for, effects of fission process on creep of, (E/T)

NITROGEN—diffusion in plutonium nitride–uranium nitride fuels, self, (E)

PLUTONIUM NITRIDES—PuN–UN, volatilization of, effects of oxygen impurity on, (E)

—PuN–UN, self-diffusion of nitrogen in, (E)

—PuN–UN, irradiation testing of fuels of

—PuN–UN, radiation effects on high temperature swelling, fission gas release, and microstructure of, (E/T)

PLUTONIUM OXIDES PuO_2—PuO_2–UO_2, radiation effects on high temperature swelling, fission gas release, and microstructure of, (E/T)

—PuO_2–UO_2, creep of, effects of fission process on, (E/T)

REACTOR FUEL ELEMENTS—development of sodium-bonded stainless steel clad plutonium nitride–uranium nitride, (E)

—performance of stainless steel clad plutonium nitride–uranium nitride, during transient irradiation, (E/T)

—swelling of, mathematical model of mechanics of

REACTOR FUELS—creep of fast, effects of fission process on, (E/T)

—radiation effects on performance of fast, at high temperature, (E/T)

REACTOR SAFETY EXPERIMENTS—fuel element irradiation in TREAT, analysis of stainless steel clad plutonium nitride–uranium nitride, (E/T)

URANIUM CARBIDES—radiation effects on high temperature swelling, fission gas release, and microstructure of, (E/T)

—fission gas diffusion and bubble growth in, (E/T)

URANIUM NITRIDES—PuN–UN, volatilization of, effects of oxygen impurity on, (E)

—PuN–UN, self-diffusion of nitrogen in, (E)

—PuN–UN, irradiation testing of fuels of

—radiation effects on high temperature swelling, fission gas release, and microstructure of, (E/T)

—PuN–UN, radiation effects on high temperature swelling, fission gas release, and microstructure of, (E/T)

URANIUM OXIDES—stoichiometry control of, (E)

URANIUM OXIDES UO_2—PuO_2–UO_2, creep of, effects of fission process on, (E/T)

—creep of, effects of fission process on, (E/T)

—radiation effects on high temperature swelling, fission gas release, and microstructure of, (E/T)

—fission gas diffusion and bubble growth in, (E/T)

—PuO_2–UO_2, radiation effects on high temperature swelling, fission gas release, and microstructure of, (E/T)

—crystal structure of, morphology and migration of lenticular voids in, (E/T) (H.D.R.)

NSA, vol. 26, 1972

263407 Klepfer, H.H. General Electric Co., Schenectady, N.Y. (USA). **Improved nuclear fuel element.** (In German). Verbessertes Kernbrennelement. **German (F.R.) patent document 2501505/A/.** Int. Cl. G21C 3-20. 14 Aug 1975. 19 p. 3 figs.

The fuel element for, e.g., a PWR or BWR contains a lining and a lubricating layer between the fuel and the fuel can. The lining consists of a material with low neutron capture cross section such as, e.g., zirconium, niobium or their alloys. It serves as a preferred reaction site for volatile impurities or fission products from the fuel. The lubricating layer of graphite or molybdenum sulphide or anorganic compositions with a laminar crystal structure reduces friction between the lining and the can. There is no direct contact between the lubricating layer and the fuel which consists of uranium or plutonium compounds, so that excessive swelling or nigh internal gas pressure due to carbon monoxide or dioxide are prevented. (DG/AK).

FUEL ELEMENTS: fission product release, protective coatings.

Atomindex, vol. 7, 1976

9843 (INIS-mf-55) TECHNICAL DEVELOPMENT OF THE FUEL RODS FABRICATION. Koll, J. H.; Biondo, C. D. (Comision Nacional de Energia Atomica, Buenos Aires (Argentina)). 1970. 14p. (In Spanish). (CONF-701124-4). INIS.

From Metallurgical meeting; Cordoba, Argentina (Nov 1970).

The development of techniques applied to the fabrication of fuel rods with core of UO_2 pellets and Zircaloy-4 (Zr-1.5% Sn) sheathing is described. For the fabrication of the fuel rods, various techniques are used; among them are powder metallurgy, machining, welding, surface treatment, and quality control. Quality control must satisfy very rigid specifications with good reproducibility because of the high cost of fabrication. The techniques described are not of exclusive nuclear application, but can be used in similar non-nuclear fabrications where equally high quality is required. (tr-auth)

NSA, vol. 26, 1972

15752 RESEARCH AND DEVELOPMENT WORK AND MANUFACTURING CAPABILITY OF ZIRCALOY TUBES FOR FUEL CLADDING FOR NUCLEAR POWER GENERATION. Kondo, Yutaka (Sumitomo Metal Industries, Ltd., Amagasaki, Japan). Karyoku Hatsuden: 22: No. 5, 528-40(May 1971). (In Japanese).

A review is presented of research and development work and manufacturing capability of Zircaloy tubes. Several products have been delivered; some are now on order from many users in Japan and abroad. Irradiation experiments carried out at Halden Reactor showed good results. Fundamental research started in 1958, followed by construction of pilot plant in 1965 and completion of production plant in 1968 is described. The present production process is composed of special melting methods, iterative vacuum melting, special heat treatment during forging, extruding with copper cladding, cold rolling with high geometrical precision, and special finishing. Present quality control is featured by strict control of composition, defects, and geometry at stages involving raw material, ingot, billet, extruded tubes, and cold rolling of tubes. Final products have to undergo rigid inspection and examination. The company is capable of providing products with various strength and ductility, by controlling content of oxygen and conditions of cold rolling and annealing. The hydride orientation index (Fn) of produced tubes is extremely small due to adoption of rolling of large thickness reduction in finishing process and removal of residual stress. Other required properties such as corrosion resistance and dimensional accuracy are realized by methods developed by the company. An ultrasonic defect detection technique with high sensitivity has been developed, and the entire circumference and length of all products are checked for defects. (Japan)

NSA, vol. 26, 1972

41356 ADVANCES IN THE TECHNOLOGY OF NUCLEAR FUEL CARBIDES AND NITRIDES. Krauth, A.; Mueller, N. (Nuklear-Chemie und -Metallurgie GmbH, Wolfgang, Ger.). pp 578-81 of Reaktortagung, Bonn, 1971. Bonn; Deutsches Atomforum E. V. (1971). (In German).

From Reactor meeting; Bonn, Germany (30 Mar 1971). See CONF-710346.

Reaction-sintering of carbide fuel pellets, or nitride fuel pellets, respectively, has proved to be a step in advancement of carbide and nitride fuel technology towards a decrease in fuel production costs. A mixture of sinterable uranium carbide powder, or uranium nitride powder is the basic material which is cold-molded and reaction-sintered either with plutonium oxide and carbon, or with a mixture of partly reacted plutonium oxide and carbon. The basic powder which represents about 80% of the total fuel material is produced by synthesis under vacuum at 1800 or 2000°C, and needs not be handled under plutonium conditions. (INIS)

NSA, vol. 26, 1972

FABRICATION, PROPERTIES, AND IRRADIATION BEHAVIOR OF U/Pu PARTICLE FUEL FOR LIGHT WATER REACTORS

HELFRID W. H. LAHR *Gelsenberg AG*
c/o Gesellschaft für Kernforschung mbH, 7500 Karlsruhe
Postfach 3640, Federal Republic of Germany

Received December 29, 1975
Accepted for Publication May 3, 1976

Advantages were offered by particle fuel fabrication using an optimized precipitation method on heavy-metal nitrate solutions from reprocessing plants. This fabrication technology (the vibration compaction of the particles in cladding tubes above all demonstrating its capability of being automated and having relatively small amounts of scrap and wastes) might result in cost savings of more than 30%.

Irradiation experiments have shown that the use of particle fuel results in advantages that can be anticipated as higher burnups and frequent load cyclings when compared with pellet fuel.

NUC. TECH., vol. 31, 1976, P. 183

1. Status Report: Licensing of Nuclear Fuel Cycle Plants, *Howard J. Larson, R. E. Cunningham (NRC)*

The Energy Reorganization Act of 1974 established the Nuclear Regulatory Commission as an independent Regulatory commission. Within the Commission, the Office of Nuclear Material Safety and Safeguards was established as one of the three statutory offices provided by the Act. The Commission, in announcing its transitory organization on January 20, 1975, provided two principal divisions within the Office: the Division of Materials and Fuel Cycle Licensing and the Division of Safeguards. Together, these divisions are responsible for the principal licensing and regulation of all facilities and materials associated with the processing, transporting, and handling of nuclear materials including safeguards to protect against threat, theft, and sabotage of certain facilities and materials. The Office is further responsible for recommending legislation necessary for the discharge of these functions by the Commission and for developing policy options for Commission consideration.

Estimated time spans from receipt of an application to licensing the startup of uranium fuel fabrication and mixed-oxide fuel fabrication plants and the construction of reprocessing plants are 10, 14, and 16 months, respectively. An environmental statement with a limited hearing for a reprocessing plant requires about 15 months. There is, however, a close interface between licensing both mixed-oxide fuel fabrication plants and reprocessing plants with pending decisions concerning use of recycled plutonium, additional safeguards options, integrated fuel cycle sites, and waste management. The industry has experienced delays in selection of sites, establishing final design, and starting construction of these plants because of ongoing studies in these areas and resolution of pending decisions. The licensing review process has also been stretched out in some instances because of these matters. Experience over the past year with intervenors concerned with licensing of reprocessing plants indicates that, in the near term, construction of certain types of fuel cycle plants are likely to be contested and licensing delays may be experienced.

Licensing casework is planned, scheduled, and monitored by the Office through its computerized project control and manpower utilization system which, in addition, periodically seeks information from applicants about their plans or changes of plans for submission of new applications. Industry participation in this scheduling program is very important for planning manpower requirements and minimizing delays in the license review process. Long-term projections for casework are made by translating projected nuclear power production to the demands it makes on the fuel cycle. The type of fuel cycle plants and the need for additional irradiated fuel storage capacity and plutonium nitrate storage capacity will depend on the timing and the type of decisions that are made with respect to widescale use of recycled plutonium and waste management.

ANS TRANS., vol. 21, 1975

39130 (A/CONF.49/P-62) STATUS OF THERMAL REACTOR FUEL MANUFACTURE IN THE UNITED STATES. Lotts, A. L.; Washburn, T. N. (Oak Ridge National Lab., Tenn.); Geller, L. (Stoller (S. M.) Corp., New York); Klepfer, H. H. (General Electric Co., Wilmington, N. C.); Layman, W. H. (Atomic Energy Commission, Washington, D. C.). Sep 1971. 16p. (CONF-710901-37). Dep. NTIS.

Prepared for fourth international conference on the peaceful uses of atomic energy; Geneva, Switzerland (6 Sep 1971).

Fuel-fabrication services are being offered in the U.S. for two classes of thermal reactors: the light-water reactor (LWR) and, less extensively, the high-temperature gas-cooled reactor (HTGR). From a fuel-fabrication requirement of approximately 1000 metric tons per year in 1970, the demand will approach 15,000 metric tons per year in 1990. Nine manufacturers are offering fuel-fabrication services for LWR's, and one firm is offering fuel for the HTGR. Although most commercial offerings are now for UO_2-fueled cores, the Pu stockpile from LWR's will stimulate the marketing of UO_2–PuO_2 fuel. Pu available for recycle will increase from a cumulative total of 2.5 metric tons in 1970 to 300 to 500 metric tons in 1990. The fuel-fabrication service normally offered includes nuclear and physical design of the fuel element and its warranty. All fuel designs in use for LWR's are based on mechanical assembly of fuel rods loaded with ceramic pellets. The principal manufacturing operations are converting UF_6 to UO_2, fabricating pellets, loading and sealing fuel rods, fabricating fuel rods into elements, and the accompanying inspection steps. The fuel element for the HTGR is a graphite block containing spherical particles coated with pyrolytic carbon and silicon carbide. In the absence of manufacturers' detailed practices and cost experience, estimates of cost trends have been made through the use of computer models of fabrication plants. The results correspond closely with quotations of the fuel manufacturers. Included in this study are the effects of such variables as plant scale, isotopes employed, fuel-element dimensions, specifications and quality-control requirements, and alternative process steps for all commercially important thermal reactor fuels. In addition, a summary of the effect of plant-expansion strategy on cost is described. 29 references. (auth)

NSA, vol. 25, 1971

1. Manufacturing Experience in the United States for Light-Water Reactor Fuel, *J. W. Lingafelter (NSC)*

INTRODUCTION

The manufacture of uranium dioxide fuel in the United States for water reactors has evolved over the past 15 years from a laboratory development engineering approach to an established manufacturing process capable of producing tonnage quantities daily. This evolution from development to full manufacturing has been based on the cumulative efforts and experience of fuel manufacturers, AEC laboratories, and AEC-funded research and development programs. A significant thrust of the evolution has been vertical process integration and process control.

Light-water reactors (LWRs) require fuel slightly enriched in ^{235}U. Enriched uranium is available in the U.S. only from the government-owned gaseous diffusion plants and only in the chemical form of uranium hexafluoride (UF_6). The manufacture of LWR fuel is carried forth in distinct, although interrelated, process operation steps. These distinct steps are:

1. conversion of UF_6 to UO_2 powder
2. conversion of UO_2 powder to sintered UO_2 pellets
3. loading and sealing of pellets into fuel rods.

A discussion of some of the more important process considerations relating to the development, manufacture, and irradiation performance of LWR fuel follows. The discussion will include the interrelationships of critical process steps in the conversion of UF_6 to UO_2 pellets, the effects of certain fuel fabrication operations on fuel irradiation performance of sintered UO_2 pellets, and the more important aspects of process and quality control.

UF_6 CONVERSION

Ceramic-grade UO_2 powder is produced in the United States primarily by using the ammonium diuranate (ADU) method. The ADU method was initially developed in the early 1960's at the Oak Ridge National Laboratory on a small laboratory basis. The Oak Ridge experience was extrapolated by most U.S. manufacturers into development and pilot-scale operations ranging up to one ton/day. In the ADU method, the starting material, uranium hexafluoride (UF_6), is converted into ceramic UO_2 powder by a series of chemical unit operations. The basic steps are vaporization, hydrolysis, precipitation, digestion, dewatering, drying, decomposition, and pulverizing. Each of these unit operations affects the ceramic quality of the resultant UO_2 powder.

Typical of the parametric effects a given operation can have on UO_2 character is that of precipitation of ammonium diuranate (ADU). Uranium dioxide (UO_2) is prepared by reacting uranium hexafluoride (UF_6) with water to hydrolyze the UF_6 and form a water solution of uranyl fluoride (UO_2F_2) and hydrofluoric acid (HF). Ammonium diuranate [$(NH_4)_2(U_2O_7)$] (or ADU) is precipitated by the addition of ammonium hydroxide (NH_4OH).

The aqueous solutions from which ADU can be precipitated may vary from neutral to acidic solutions. The quantity of ammonium hydroxide used to precipitate ammonium diuranate from aqueous uranyl fluoride may range from 1 to 3 times the stoichiometric quantity.

(Cont'd pg. 19)

Temperature is known to have an effect on the initial precipitation. Suitably effective temperature should be controlled within ±5°C at or around 25°C. Other important interrelated parameters in ADU precipitation are the effects of recycle lines and digestion time and temperature. Each of these is known to influence the character of the ADU precipitate and, in turn, the character of the resultant UO_2 powder.

UO_2 PELLETIZING AND SINTERING

The ceramic-grade UO_2 powder is processed into dense pellets using ceramic processing techniques. These techniques include compaction, granulation, sizing pelletizing, and sintering. Demanding production quality requirements are placed on the final sintered UO_2 pellet in terms of density, dimension, and integrity. All three of these product characteristics are influenced by the ceramic processing steps.

Typical of process operations in the pelletizing and sintering step is the pelletizing operation, which is perhaps the most important, as well as parametrically sensitive, operation. The goal of the pelletizing operation is to produce green pellets that will sinter to uniform density, diameter, and length, on a high volume production basis. Obstacles in obtaining that goal are variations in:

1. intrinsic particle shape and size of the UO_2
2. bulk density of feed UO_2
3. press tooling
4. press compaction, dwell, and ejection pressures.

Modern hydraulic presses have been developed where there are sufficient controls to accommodate a wide variation in the fuel UO_2. Unfortunately, this accommodation is usually at the sacrifice of production rate.

FUEL ROD ASSEMBLY

Pellets, cladding, and end plugs are brought together in the fuel rod assembly step. Considerable development effort has been expended in the production of Zircaloy cladding for fuel rods, and in the welding of end plugs to the cladding.

Use of the tungsten inert gas (TIG) welding process with helium as a shielding gas has been standard. Boxes or chambers used to provide a controlled protective environment for welding have gone through considerable development. Presently, the first end plug is welded to the tubing end using a single shielding box to exclude the atmosphere. The dried pellets are loaded into the tube with one end welded. The loaded tube is then placed into a weld chamber, the air atmosphere evacuated, and then back-filled with helium. The final end-plug weld is made in the chamber. With this technique, leak-tight and radiographically acceptable welds are produced consistently.

FUEL FABRICATION AND IRRADIATION PERFORMANCE

Fuel irradiation experience has shown that fuel fabrication operations can affect the lifetime performance of fuel. These operations have been established either as a result of using processing sequences based on the ceramic industry, or in response to specific pellet product specifications. Examples of pellet fabrication operations that have been shown to affect fuel performance are: pore size control of the green pellet, density of the sintered pellet, powder binders and lubricants, and grinding and drying of sintered pellets.

Lubricating materials have been added to UO_2 powders in an attempt to provide more uniform flow of the powders into the die cavity. Some materials have been added to increase the green strength of pellets prior to sintering or to agglomerate the individual powder grains to effect more uniform flow and die fill; i.e., binders. Other materials have been added to powders to enhance the rate of pellet sintering and achieve early desired final density.

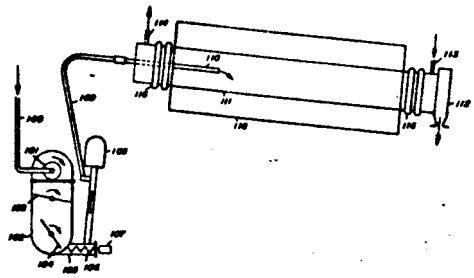
If the processing operations do not completely remove the organic waxes and oils, these remain within pores of the sintered pellets. These hydrocarbons can thermally decompose during irradiation and release the hydrogen to the fuel rod when the UO_2 pellets crack from thermal stresses during irradiation. The thermally decomposed (dissociated) hydrocarbons have been shown to be a source of atomic and molecular hydrogen which can react with the Zircaloy cladding. The formation of ZrH_2 zones (nuggets) in the cladding can breach the cladding and cause fuel rod failure.

The sintered UO_2 pellets are generally centerless ground to obtain the desired diameter tolerance, and are subsequently dried. In the 1960's, moisture content of pellets loaded into fuel rods was 30 to 40 ppm. Other volatile content of pellets generally was not specified. Fuel irradiation experience subsequently showed that cladding hydriding or cracking can result when moisture content is in the range of 50 to 75 ppm. As a result of this irradiation experience, changes in the fuel manufacturing steps were instituted. Most U.S. fuel manufacturers now vacuum outgas the ground pellets to limit moisture in the pellet to 10 ppm and other volatile content to 5 ppm. Control of moisture in pellets must also consider effect of moisture pickup of pellets from the atmosphere after outgassing.

One very recent fuel irradiation phenomenon has been the densification of low-density pellets during irradiation. The low-density problem is a direct result of changing process specifications, and in turn, pellet manufacturing techniques, to meet product specifications. The ceramic technology used to produce pellet porosity in low-density pellets could not mitigate against further sintering of the pellet during irradiation. The irradiation-induced sintering produced pellet shrinkage, and in turn, cladding failure.

From a metallurgical and mechanical properties viewpoint, the fabrication of Zircaloy cladding has an effect on irradiation performance. Percent cold work, grain size, hydride orientation, and texture of the cladding are important variables. These variables, if not properly controlled, can lead to bowing or warping of the cladding, or to cladding failures, during fuel irradiation.

3,734,694
APPARATUS FOR PRODUCING UO_2 POWDER
John J. McCoy, San Jose, Calif., assignor to General Electric Company
Original application Apr. 25, 1968, Ser. No. 724,119, now Patent No. 3,579,311. Divided and this application Dec. 2, 1969, Ser. No. 881,438
Int. Cl. C01g *43/02*
U.S. Cl. 23—262 **3 Claims**



An improved apparatus for preparing uranium dioxide powder suitable for use in the fabrication of fuel for nuclear reactors. The apparatus has a hopper for receiving dewatered sludge, mixing means within the hopper to maintain a uniform slurry in the hopper, means for receiving the slurry from the hopper and feeding the slurry through at least one outlet, and a rotating tube calciner adapted to be maintained at a temperature in excess of 1000° F. with the outlet positioned at about the center of the calciner circumference adjacent one end thereof.

U.S. PATENTS 1974

24672 (DP-MS-72-74) DEVELOPMENT OF HIGH PERFORMANCE URANIUM METAL FUELS FOR SAVANNAH RIVER REACTORS. McDonell, W. R.; Sturcken, E. F. (Du Pont de Nemours (E.I.) and Co., Aiken, S.C. (USA). Savannah River Lab.). 1973. Contract AT(07-2)-1. 32p. (CONF-731101-52). Dep. NTIS $3.75.

From joint meeting of the American Nuclear Society and the Atomic Industrial Forum and Nuclear Energy Exhibition; San Francisco, California, USA (11 Nov 1973).

The technological evolution of high performance uranium metal fuel elements for Savannah River Plant (SRP) reactors is described, with emphasis on the metallurgical development required to achieve satisfactory irradiation behavior. Fuel element design changes, embodying principally increased heat transfer area, were accompanied by improved methods for bonding the uranium cores within aluminum cans. Anisotropic growth of the fuel element during irradiation caused by texture of the core was eliminated by development of beta heat-treating processes using oil-quenching. Cavitational swelling was controlled by minor alloying additions to the uranium metal. The resulting fuel elements are capable of sustaining high exposures in SRP reactors. (auth)

NSA, vol. 29, 1974

4905 NUCLEAR FUEL ELEMENT. Maki, Hideo. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1974-19758/B/. 28 Dec 1968. 5p. (In Japanese).

In conventional nuclear fuel elements, a single space is provided within a sheath tube and a spring is disposed therein, thus intending to expand and move fuel pellets in the sheath tube along its length during reactor operation. Nevertheless, the fuel pellets do not always move freely on account of metallographic or density variations in the fuel pellet as well as cubic expansion attributable to the fission products. Therefore, the sheath tube is stressed by the fuel pellets and is sometimes damaged. Even in cases where the fuel pellets can expand and move, changes in the fuel distribution in the reactor core arise when the movement is made in only one direction. It is therefore feared that the nuclear properties will suffer. According to this invention, at least two spaces are provided within the sheath tube and the springs are disposed in these spaces, so that expansion and movement of the fuel pellets can take place in two directions. Herein, if all the spaces can absorb lengths exceeding the amounts of expansion of the fuel pellets, the lower fuel pellets will considerably descend due to their weight, nonuniformity of expansion, etc., and hence, the nuclear properties will suffer. In this respect, according to this invention, the springs are set so that the lower space may reach the limit of absorption earlier than the upper space. (JA)

NSA, vol. 33, 1976

37836 PRESSURE EQUALIZATION SYSTEM FOR FUEL PINS OF WATER-COOLED REACTORS. Malang, S.; Mueller, St. (Kernforschungszentrum, Karlsruhe, Ger.). Kerntechnik; 14: No. 3, 114-17(Mar 1972). (In German and English).

A pressure equalization system was developed for fuel pins of water-cooled reactors, which makes possible a major reduction of mechanical stresses in the cladding tubes. In boiling water reactors, a complete equalization of internal and external pressure can be achieved over the entire service life of the pins. This prevents a creep collapse of the cladding onto the fuel, and avoids any exchange between the coolant and fission gases in the event of small leak defects. Satisfactory functioning of the system was demonstrated in out-of-pile tests. (auth)

NSA, vol. 26, 1972

35182 FABRICATION TECHNIQUE OF JOYO FUEL. Matsumoto, K.; Ito, I.; Sasao, N.; Sakamoto, K. (Power Reactor and Nuclear Fuel Development Corp., Tokai, Ibaraki (Japan). Tokai Works). Doryokuro Giho; 45-61(Mar 1974). (In Japanese).

86 fuel assemblies of JOYO's initial core are being fabricated and inspected at PNC's fuel fabrication factory. Technical aspects of fabrication and inspection are reported. The manufacture of PuO_2-UO_2 mixed oxide pellets is reviewed. Specifications, fabrication, and inspection are explained. Results of the statistical analysis on pellet diameter, density, and distortion are shown. The fabrication of fuel pins is reviewed. The process is explained with a flow sheet. The techniques for the welding of end plugs and the winding of spacer wire are discussed together with various results of measurements and statistical analysis. The methods and results of various strength tests performed on welded parts are also discussed. The fabrication of fuel assemblies is discussed with emphasis on welding technique. The techniques of fuel recovery and the disposal of waste liquid are discussed. The dry and wet recovery processes are explained with flow sheets and photographs of equipments. (JA)

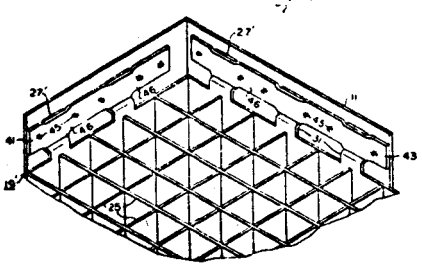
NSA, vol. 31, 1975

1118 PROCESS FOR THE PRODUCTION OF NUCLEAR FUEL FOR NUCLEAR REACTORS. Marlowe, M. O. (to General Electric Co., Schenectady, N. Y. (USA)). Swiss Patent Document 539924/D/. 14 Sep 1973. 4p. (In German).

A procedure is described for reducing the viscosity and creep resistance, at the operating temperature of the reactor, of nuclear fuel containing at least one compound of a fissile isotope. An additive chosen from the group consisting of gadolinium oxide, cesium dioxide, yttrium oxide, W, Mo, and their mixtures is incorporated in the fuel. The quantity added (from 0.1 to 5 wt. %) is sufficient to decrease the viscosity of the fuel and allow its plastic creep. (tr-INIS)

NSA, vol. 33, 1976

3,719,560
FUEL ASSEMBLY FOR A NUCLEAR REACTOR USING ZIRCONIUM ALLOY CLAD FUEL RODS
Joseph B. Mayers, Greensburg, and Stanley Kmonk, Pittsburgh, both of Pa., assignors to Westinghouse Electric Corporation, Pittsburgh, Pa.
Filed April 29, 1969, Ser. No. 820,107
Int. Cl. G21c 3/34
U.S. Cl. 176—78 13 Claims



Zirconium alloy structural materials for fuel rods are desirable in nuclear fuel assembly construction due to their low neutron absorption cross section properties. However, zirconium alloys are difficult to utilize due to their inability to be welded with other commonly-used fuel assembly construction materials such as inconel or stainless steel and their thermal expansion coefficient, which is significantly smaller than the inconel or stainless steel alloys commonly-used. The latter characteristic may result in buckling of the fuel assembly in a changing thermal environment. To obviate these difficulties in utilizing fuel rod cladding and/or fuel assembly cans of zirconium alloy with other structural elements such as a spacer grid formed from other materials, provisions are made herein for: (1) a fuel assembly construction having a zirconium alloy enclosure including means for attaching structural elements such as nozzles or grids of other materials thereto, and (2) a stainless steel enclosed fuel assembly having zirconium alloy clad fuel rods, including means for permitting relative movement of the fuel assembly parts due to differences in the thermal expansion rates of elements in the fuel assembly.

U.S. PATENTS 1973

2. Standards Program for the Nuclear Fuel Cycle, *Robert B. Minogue (NRC)*

This paper presents a report on some aspects of the national nuclear standards program related to developing rules and guides that define acceptable methods for resolving some of the major problem areas in the nuclear fuel cycle. This program is being carried out by the technical staff of the Nuclear Regulatory Commission (NRC) and with substantial input from and use of standards developed within the national consensus standards program coordinated by the American National Standards Institute (ANSI).

Our objective is to identify and promulgate solutions to generic issues of safeguards, siting, and safety associated with regulation of nuclear fuel cycle facilities and materials. The basic safety standards that set policy on acceptable levels of risk are mostly incorporated in the Code of Federal Regulations, under Title 10. In addition to the Regulations, there are the Regulatory Guides developed by the NRC staff. Certain of these guides indicate to applicants the information that should be submitted in support of license applications and form part of the basis of licensing decisions. Other guides provide information on techniques used by the Regulatory staff in evaluating safety problems. Finally, there are the important Regulatory Guides that identify good engineering practice; these are the guides in which ideally we use the product of the national consensus standards program to define how the required level of performance is to be achieved.

A major potential problem area in the nuclear fuel cycle relates to the disposition or reuse of plutonium produced in power reactors. We are developing improved techniques for control and accountability of special nuclear materials at fixed sites and in transit and upgrading physical security requirements for plants and materials. The toxicity of plutonium and the other transuranics requires careful consideration in developing the technological bases for siting facilities. A study is under way to systematically assess this problem to provide a basis for future siting regulations, standards, and guides.

Other important areas of standards needs are related to the appropriate level of application of seismic criteria and seismic analysis methods to fuel reprocessing and mixed-oxide fuel fabrication plants. Engineering studies and assessment of the applicability of existing standards and techniques have been initiated to provide a basis for these standards. Decommissioning criteria are needed for all fuel cycle facilities. Establishment of numerical guidelines for design objectives and limiting conditions for operation to meet "as-low-as-practicable" criteria for effluents from fuel cycle facilities is another problem area that is being dealt with by our technical staff. Standardized methods of effluent monitoring are being developed.

Existing Regulatory programs and standards for shipping containers are being continuously evaluated, modi-

(Cont'd pg. 22)

fied, and upgraded as appropriate to assure a high level of protection to the public during transportation of increasing numbers and sizes of radioactive material packages.

Studies will be initiated to provide bases for rules and guides relating to siting, design, construction, and operation of waste storage facilities, structural requirements for containers and shipping casks, and other aspects of waste management.

In summary, I have identified some of the major problem areas in the nuclear fuel cycle and the related plans in the nuclear standards development program. Resolution of the issues involved and the timely development of the needed standards will require the combined efforts of the NRC technical staff and the ANSI national standards groups.

ANS **TRANS., vol. 21, 1975**

20731 RESEARCH AND DEVELOPMENT OF JAPANESE PRODUCTION OF NUCLEAR FUEL. **Mishima, Y. (Tokyo Univ.). Genshiryoku Kogyo; 17: No. 9, 8-11(Sep 1971). (In Japanese).**

Research and development progress on nuclear fuel has been made in Japan for about 15 years. Fuel has been produced for research reactors, however, manufacture for commercial power plants has been limited. For domestic production of the nuclear fuel, close coordination between user utility companies and manufacturers is essential. The research, development, and manufacture during the past 15 years are reviewed including efforts devoted to fuel and structural materials by educational institutions, research institutes, and private enterprise. Activity in Nuclear Fuel Safety Committee of Japan Atomic Industrial Forum, dealing mainly with zirconium and stainless steel cladding tubes, is discussed. The joint studies in the committee contributed greatly to establishment of manufacturing techniques in Japan. (Japan)

NSA, vol. 26, 1972

6. Soluble Poison for Nuclear Safety Control During Fuel Processing/ *W. G. Morrison, J. A. Buckham, A. L. Ayers (PPC)*

Soluble neutron poison has been used in a pioneering achievement at the Idaho Chemical Processing Plant as the primary nuclear safety control for routine processing of irradiated reactor fuel. This innovation has permitted safe use of large nongeometrically-safe headend equipment for semicontinuous processing of enriched uranium-zirconium fuel with a resultant increase in the reprocessing capacity of approximately tenfold.

Large vessels in the original zirconium fuel process[1] headend were installed for batch-type operation, and capacity was severely limited for criticality reasons. Impending increased fuel loads necessitated process changes to increase capacity significantly. Preliminary investigations indicated that adequate capacity could be obtained safely with the existing headend equipment essentially unchanged at modest expense by use of reasonable concentrations of boron poison in the process solutions. Extensive neutron diffusion calculations demonstrated conclusively the safety of the process with this approach under normal and all foreseeable abnormal conditions of operation. The metal volume fraction (MVF) concept was used in the calculations rather than arbitrarily assumed fuel configurations because of the unknown instantaneous conditions inside of a dissolver during a dissolution process. With the type of fuel to be processed, these calculations showed that a 0.6 MVF was the most reactive condition. All subsequent calculations considered fuel present at 0.6 MVF to simulate the maximum reactivity that could result from any possible fuel configuration.

For this condition together with complete precipitation and settling of uranium from the dissolver solution, which is about the most severe abnormal operation situation conceivable, a natural boron concentration of 3.8 g/liter in the dissolvent was calculated to give a k_{eff} of 0.9 for the dissolver. This boron concentration was selected for plant use. Under all expected normal operating conditions with this boron concentration, the calculated k_{eff} value is under 0.8. Included in the abnormal situations investigated for 3.8 g boron in the dissolvent were a) rupture of the fuel basket with undissolved fuel accumulating in the annulus which gave a k_{eff} of 0.96, and b) plateout of $UF_4 \cdot 3/4H_2O$ on the inside surfaces of the dissolver which showed that an unusually large amount of U-235 (90 to 100 kg) is necessary to present a hazard. A k_{eff} of 0.96 was not considered unreasonably high in view of the severe assumed conditions which included almost 100% excess U-235 over the normal charge present at the time of rupture. The computational methods have been used to correlate measured results from clean critical experiments[2] performed with boron poisoned arrays of fuel elements at the ORNL Critical Mass Laboratory. Very good agreement between calculated and measured results was found[3].

Prestartup neutron multiplication measurements in the dissolver with borated water showed that the neutron multiplication did not exceed 2.0 with up to 9.9 kg U-235 arranged at 0.6 MVF for maximum reactivity. Also, no significant change in the neutron count rate was measured during the course of several dissolution tests with unirradiated enriched fuel.

During operation, suitable safeguards to maintain nuclear safety include dual neutron attenuation measurement instruments with an automatic shutoff in the dissolvent feed line, use of a closed-loop borated cooling-water system for heat removal, and the absence of any water connections to the facility[4]. In addition, use of carefully prepared standard operating procedures under strict administrative control insures effective control of nuclear safety through the use of soluble poison in this fuel processing facility. No problems have arisen related to use of the soluble poison during extended periods of operation in 1964-66.

---

1. REID, D. G., C. E. STEVENSON, R. B. LEMON, and F. K. WRIGLEY, *Proc. Second U.N. Intern. Conf. Peaceful Uses At. Energy*, 17, 145 (1958).
2. JOHNSON, E. B. and R. K. REEDY, Jr., "Critical Experiments with SPERT-D Fuel Elements," ORNL-TM-1207 (1965).
3. FOX, J. K., "Comparison of ORNL SPERT-D Critical Experiments with Calculations," IDO-17093 (1965).
4. BOWER, J. R., "Zirconium Processing Capability of the Idaho Processing Plant," IDO-14645 (1964).

ANS TRANS., vol. 9, 1966

36514 KINETICS OF SINTERING OF UO_2. Moorthy, V. K.; Soni, N. C. (Bhabha Atomic Research Center, Bombay). Trans. Indian Ceram. Soc.; 27: No. 6, 177-88(1968).

The kinetics of sintering of pellets prepared from UO_2 powders derived from ammonium diuranate at different calcination temperature were studied. The powders were calcined between 500 and 1500°C and the densification characteristics of the pellets made from these were evaluated for different soaking times from 10 to 1000 min at sintering temperatures between 1000 and 1500°C. Data were analyzed using the empirical equation D = C + K log Th t where D is the density, C and K are constants, and t denotes time. It was found that the progress of sintering, attainment of limiting densities, and variation in rates of densification of different temperatures and time can be explained on the assumption that material movement (hence densification) is governed by the number, nature, and energy of crystalline defects (flaws) in the original powders. (Indian Sci. Abstr.)

NSA, vol. 26, 1972

261027 Narita, Daisuke; Imai, Tadamitsu; Masuda, Sumio; Naruki, Kaoru; Koizumi, Masumichi (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Test fabrication of mixed oxide pellet for light water reactor.
Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan). Semi-annual progress report of Power Reactor and Nuclear Fuel Development Corporation, Tokai Works, Jan.-Jun., 1974. PNCT-831-74-02. Dec 1974. p. 59-67.

The in-pile densification of LWR fuel gives rise to the safety problem of reactors. For the 95% T.D. pellets of mixed 4 wt% PuO_2 - 96 wt% UO_2, the following matters were studied : the effects of powder characteristics, compacting pressure and sintering temperature on pellet density; the effects of sintering time and sintering temperature on the crystalline grain growth; and plutonium homogeneity of the pellets. (1) The particle size of powder corresponds to the density of sintered pellets. The particle size smaller than 0.5 micron is optimal for the target density of 95%. (2) Powder compacting pressure has little influence on the density of sintered pellets. (3) Sintering temperature influences largely the grain growth of pellets, while the sintering time does little. (Mori, K.).
FUEL PELLETS: sintering; SINTERING: density.

Atomindex, vol. 7, 1976

33847 (PNC-N-831-70-2, pp 11-14) PREPARATION OF URANIUM CARBIDE AND ITS PELLET FABRICATION. Naruki, Kaoru; Narita, Daisuke; Takanobu, Osamu; Muto, Tadashi (Power Reactor and Nuclear Fuel Development Corp., Tokai (Japan)). Dec 1970. (CONF-700345-1).

From Annual meeting of the Atomic Energy Society of Japan; Hiratsuka, Japan (27 Mar 1970).

Efforts for the preparation of carbide fuels were directed to develop carbothermic reduction, cold pressing, and sintering methods to prepare uranium carbide pellets. The effect of binder for consolidation of UO_2-C powder blend was investigated. The PVA binder resulted in an increase of carbon content of the carbide. This may be due to carbon contribution from the binder. Crushing and extended heat treatment at low temperature resulted in removal of the binder from pellets. Other binders were compared on fabricability of green pellets. It was difficult to tell superiority between them from their sintered density. Experiments were conducted on comminution of carbide granules as prepared by the carbothermic reduction. The experiment revealed that wet ball milling with CCl_4 was more effective than dry ball milling when a vibratory mill was used. The wet ball milled powder gave higher sintered density in spite of a little larger particle size than the dry ball milled powder. (Japan)

NSA, vol. 26, 1972

32027 (A/CONF.28/P/233) OXIDE FUEL FABRICATION AND PERFORMANCE. S. Naymark (General Electric Co. [Atomic Power Equipment Dept.], San Jose, Calif.) and C. N. Spalaris. 15p.

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

Design considerations and fabrication techniques are surveyed for oxide fuels used in power water-cooled reactors. Performance of Zircaloy-2- and stainless-steel-clad fuel elements is reviewed. Fission gas release from fuel elements is considered. Possible future improvements in the fuel elements are discussed. (D.L.C.)

NSA, vol. 18, 1964

44762 (ORNL-4560, pp 218-34) DEVELOPMENT OF FUEL ELEMENT FABRICATION. (Oak Ridge National Lab., Tenn.).

From fuels and materials development program quarterly progress report for period ending March 31, 1970.

Studies of the irradiation performance of Al-base dispersion fuel elements are continuing along with an investigation of improved processing techniques for the fabrication of dispersion plates. Examination of 50% high-fired U_3O_8-Al dispersions irradiated to an average fission density of 2.24×10^{21} fissions/cm^3 is reported. Evidence of incipient failure was observed at the end of this plate, where the burnup corresponded to 2.5×10^{21} fissions/cm^3. Dispersion plates that contain the burned grade of U_3O_8 appear to be more resistant to blistering after irradiation than those that contain the high-fired grade of oxide. Also, the presence of fines increases the temperature at which the plates that contain the high-fired grade of U_3O_8 blister after irradiation. Investigations of the use of scattered and fluorescent radiation for measuring the thickness of cladding were continued along with studies of the angle, intensity, and energy of scattered radiation. (auth)

NSA, vol. 24, 1970

48372 (HW-76347) OBSERVATIONS ON PRODUCING DUCTILE ZIRCALOY-2 STRIP AND SHEET. O'Keefe, D. P. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). 11 Jan 1963. 7p. Dep. NTIS.

Procedures used successfully for the production of Zircaloy-2 strip and sheet in small lots are described. Fabrication procedures with extrusion as the primary reduction from ingot followed by rolling have been found to produce very ductile sheet. The extrusion method controls texturing, which affects bend ductility. Heterogeneous microstructure is to be avoided because of fatigue strength requirements and because of orange peeling that will result if the zone of critical strain is near the surface. (P.C.H.)

NSA, vol. 26, 1972

30915 (ARH-2334) SELECTIVE DISSOLUTION OF URANIUM FROM FUEL FABRICATION SCRAP. Panesko, J. Vincent (Atlantic Richfield Hanford Co., Richland, Wash.). Mar 1972. Contract AT(45-1)-2130. 10p. Dep. NTIS.

Uranium was preferentially dissolved from unsintered plutonium-uranium fuel fabrication scraps. Trace levels of dissolved plutonium were recovered by anion exchange. These processes were not applicable to sintered scrap containing solid solutions of PuO_2-UO_2 in which plutonium dissolved with uranium. (auth)

NSA, vol. 26, 1972

3,898,309

METHOD OF FORMING HIGH DENSITY OXIDE PELLETS BY HOT PRESSING AT 50°-100°C ABOVE THE CUBIC TO MONOCLINIC PHASE TRANSFORMATION TEMPERATURE

Arvid E. Pasto, Oak Ridge, Tenn., assignor to The United States of America as represented by the United States Energy Research and Development Administration, Washington, D.C.

Filed May 11, 1973, Ser. No. 359,397
Int. Cl.[2] G21C *21/00*
U.S. Cl. 264—.5 10 Claims

1. In a method of hot pressing a rare earth or actinide oxide having a cubic to monoclinic phase transformation wherein powders of said oxide are placed in a die and heated to a sintering temperature and sintered under pressure; the improvement comprising the steps of;

heating said oxide in the cubic form to a temperature within the range of 50° to 100°C above said transformation temperature, and,

maintining the temperature within said range while said oxide is under pressure until said oxide has sintered to substantially theoretical density.

10. A sintered compact of monoclinic Eu_2O_3 having a density of about 100% theoretical density, a purity of 99.9 wt. %, and a grain size of less than 100 microns.

U.S. PATENTS 1973

17133 MANUFACTURE OF NUCLEAR FUEL ELEMENTS. Pollet, M.; Baschwitz, R.; Serole, B.; Thiebaut, B. (Compagnie pour l'Etude et la Realisation de Combustibles Atomiqaues, Paris). Kerntech. Atomprax.; 13: No. 11, 488-90(Nov 1971).

The manufacture of laminated fuel elements containing specially sintered fuels and clad by resisting light alloys, the manufacture of ceramic fuel elements for pressurized-water reactors, and the manufacture of coated particles for high-temperature reactors are described. (auth)

NSA, vol. 26, 1972

20867 (CNLM-1626) FABRICATION AND INSPECTION REPORT FOR CERAMIC TYPE FUEL ELEMENT. (Pratt and Whitney Aircraft, Middletown, Conn. (USA). Connecticut Advanced Nuclear Engineering Lab.). 7 Jan 1959. 55p. Dep. NTIS $4.50.

Declassified 9 Oct 1971.

Data and information are presented on fabrication and inspection of ceramic $BeO-UO_2$ fuel elements. The data and information are presented on sheets listing fabrication history and inspection history. (JRD)

NSA, vol. 33, 1976

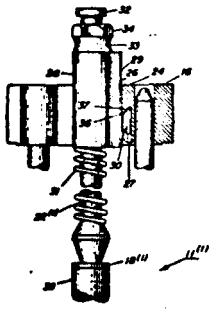
3,741,868

FUEL BUNDLE WITH REMOVABLE RODS

Frank D. Qurnell, Robert N. Ikemoto, and James L. Lass, San Jose, Calif., assignors to General Electric Company

Filed Mar. 30, 1970, Ser. No. 23,723
Int. Cl. G21c *3/34*
U.S. Cl. 176—78 26 Claims

A removable rod arrangement for the fuel bundle of a nuclear reactor core providing ready insertion and removal of the removable rods without disassembly of the fuel bundle wherein the removable rods are accurately positioned by complementary positioning surfaces and retained in position by a spring-loaded, tapered pin and pin seat.



U.S. PATENTS 1973

43410 (BARC—596) STUDIES ON ELECTRICAL CONDUCTIVITY OF REFIRED SINTERED URANIUM DIOXIDE PELLETS. Rao, S. V. K.; Agnihotri, P. K.; Anandan, N. S.; Prasad, G. E.; Moorthy, V. K. (Bhabha Atomic Research Centre, Bombay (India)). 1972. 11p. Dep. NTIS (U. S. Sales Only).

In earlier investigations, it was reported that the electrical conductivity of sintered uranium dioxide pellets depended on the calcination temperature and the sintering treatments and could be related to the microstructure attained. The highest sintering temperature studied was 1700°C. High-density pellets prepared by sintering at 1500°C were refired at 1800, 1900, and 2000°C. Though there were no significant changes in the densities, the electrical conductivity of the samples varied considerably, showing a minima at a refiring temperature of 1900°C. Attempts were made to correlate the changes in electrical conductivity to the microstructural changes which still seem to be influenced by the thermal history of the samples. (auth)

NSA, vol. 26, 1972

31441 SINTERING STUDIES ON UO_2: CHANGES IN ELECTRICAL CONDUCTIVITY WITH THE PROGRESS OF SINTERING. Rao, S. V. K.; Moorthy, V. K. (Bhabha Atomic Research Centre, Trombay, India). pp 119-40 of Proceedings of the Symposium on Powders and Sintered Products. Bombay; Dept. of Atomic Energy (1971).

From Symposium on powders and sintered products; Kanpur, India (31 Jan 1971). See CONF-710118.

UO_2 powders differing in their mode of origin and calcination temperature differ widely in their sintering behavior. It was observed that the poorer the crystallinity in the resultant UO_2 powders the greater was their sinterability. The sintering of UO_2 powder compacts brings about changes in crystallinity, density, and microstructure. It was considered that such changes during the progress of sintering would be reflected in changes of electric conductivity of the UO_2 pellets. Specimens of UO_2 pellets were made by cold-pressing powders of different crystallinity prepared from ADU by calcination/reduction at temperatures from 600 to 1500°C and sintered at temperatures of 600 to 1500°C. Samples from some of the powders were sintered at temperatures up to 1700°C. Electric conductivities of all the specimens were measured at room temperature. Attempts were made to explain the changes in conductivity on the basis of crystallinity development in the powders, densification of the compacts, and changes in microstructure. (auth)

NSA, vol. 26, 1972

41933 (TID-11295(3rd Ed.)) NUCLEAR FUELS AND MATERIALS DEVELOPMENT. William L. R. Rice, ed. (Division of Reactor Development, AEC). July 1964. 304p.

A summary is given of the work done in 1963 under the Advanced Reactor Technology Program of the Division of Reactor Development. Several U. S.–Euratom joint projects are also described. The materials and topics covered include fuel and fertile materials, cladding and structural materials, moderator materials, and materials testing. (D.L.C.)

NSA, vol. 18, 1964

CONTROLLED POROSITY OXIDE FUEL

H. ROGAN T. J. HEAL J. E. LITTLECHILD and L. RAVEN

British Nuclear Fuels Limited. Springfields Works

SUMMARY

The use of the concept of controlled porosity to tailor the microstructure of UO_2 fuel pieces to optimise fuel performance under irradiation is described and reference is made to experience during the large-scale manufacture of fuel of this type using the Integrated Dry Route (IDR) for conversion of UF_6 to UO_2. The experience is drawn from both binder and binderless pelleting routes. The parameters necessary to define and control the quality of controlled porosity fuel are described and irradiation experience is outlined, with particular reference to the performance of BNFL controlled porosity fuel in water reactor irradiation experiments.

NUC. EN. MATUR., vol. 7, 1976

3,867,489
METHOD OF PRODUCING AN OXIDATION-RESISTANT UO_2 FUEL BODY

Jack A. Rubin, Encino, Calif., assignor to The United States of America as represented by the United States Atomic Energy Commission, Washington, D.C.

Filed July 17, 1962, Ser. No. 210,880

Int. Cl. G21c *20/00*

U.S. Cl. 264—0.5 9 Claims

1. A method of forming a UO_2 fuel body resistive to oxidative attack, which comprises mixing UO_2 and a ceramic oxide of relatively low thermal neutron absorption cross section, adding to the resulting mixture a small amount of a compound selected from the class consisting of alkali metal halides and alkaline earth metal halides, and then forming a fuel body of the resulting composition.

U.S. PATENTS 1975

9841 (AE-415) INFLUENCE OF POWER CHARACTERISTICS ON PROCESS AND PRODUCT PARAMETERS IN UO_2 PELLETIZATION. Runfors, U. (Aktiebolaget Atomenergi, Studsvik (Sweden)). Apr 1971. 33p. Dep. NTIS (U. S. Sales Only).

The relations between the powder properties and the different steps in the pelletizing process as applied to uranium dioxide were examined. The results showed that, no simple correlations exist between powder properties and the properties of the final product. Powder metallurgy or the entire pelletizing process can be understood, if the influence of powder properties on the partial processes of powder metallurgy and on the intermediate products are clarified and then integrated. This can be achieved if process variables as well as powder and product porperties are chosen in a meaningful way and if a sufficient number of powder properties are kept constant, which is difficult in powder metallurgy. This integrated approach contains the possibility of studying powder metallurgy as an optimization problem, technically as well as economically. (auth) (Sweden)

NSA, vol. 26, 1972

17590 NUCLEAR FUEL DESIGN FOR LIGHT WATER REACTORS. Sawaguti, Y. (Tokyo Electric Power Co., Inc.). Genshiryoku Kogyo; 19: No. 2, 14-19(Feb 1973). (In Japanese).

History, present problems and future trend of the fuel design for light water reactors (BWRs constructed by GE and PWRs constructed by WH) are reviewed. Change of the fuel design for BWRs from BWR/1 to BWR/5 is explained from the view points of cladding material, length of fuel rods, number of fuel rods per assembly, shape of pellets, and design criteria. Power density (50.8 kW/l) and linear heat rate (18.5 kW/ft) of BWR/4 and 5 are appreciated as the results of improved design. As the present problem, cause of fuel rupture and the counter measures are shortly discussed. As the future trend, improved design of BWR/6 is explained. The increased power density due to the improvement of local power evenness associated with the employment of 8 × 8 rod (148 inches long) assembly and the decreased linear heat rate and fuel temperature are main features of the future BWRs. As for the PWRs, change of cladding material (from stainless steel to Zr-4), size of subassembly, type of control rods, effective length of fuel rods and use of part-length control rods are pointed out as the causes of improved power density and neutron economy. Deformation of cladding tubes due to outer pressure and its counter-measures are explained as the present problem. Like BWRs, decrease of the linear heat rate and fuel temperature is feature of future PWR design. (JA)

NSA, vol. 30, 1974

48371 (CISE-R-292) CHARACTERIZATION AND CONTROL OF SOME URANIUM TUBES COEXTRUDED WITH ZIRCALOY CLADDING. Scaroni, A. (Centro Informazioni Studi Esperienze, Milan (Italy)). Dec 1969. 71p. INIS.

Declassified 1971.

The activities performed on the fabrication and evaluation of some metallic uranium fuel tubes with Zircaloy-2 cladding coextruded by Nuclear Metals Inc. on the basis of CISE specifications are described. The work is a basic part of the experimental program, sponsored by EURATOM and CNEN, for the development of the metallic CIRENE fuel element. After the description of the fabrication process, an account is given of the measurements and tests carried out at NMI, at SORIN, and at CISE for the control of the initial specifications and for the characterization of the products. Some final notes, concerning both the work described and other relevant activities, are added. (INIS)

NSA, vol. 26, 1972

14678 NUCLEAR FUEL CYCLE: WHAT'S HAPPENING TODAY. Schwieger, R. G. Power; 117: No. 9, 29-36; 112-115(Sep 1973).

The growth of nuclear power production is briefly discussed. Licensing delays, soaring plant costs, and fuel failures have led to a different philosophy on fuel management and more interest is placed on enrichment with price advances for services and the proposed changes in contractual arrangements with the government-owned diffusion plants. Utilities and fuel manufacturers are investigating plutonium recycle and the subsequent savings in fuel-cycle cost that it offers. Gas-centrifuge is compared with gaseous diffusion for producing enriched uranium. A pooling or brokerage service is necessary for separative work units due to AEC's new fixed-commitment contracts. Licensing, fuel design, fuel manufacturing, and the uranium market are discussed. (MCW)

NSA, vol. 30, 1974

10270 (ORNL-TM-2357) FUELS FOR WATER REACTORS. Scott, J. L.; Weaver, S. C. (Oak Ridge National Lab., Tenn.). Dec. 1968. Contract W-7405-eng-26. 25p. Dep. CFSTI.

Uranium dioxide (UO_2) is universally selected as fuel for water reactors because of its superior compatibility with stainless steel or Zircaloy cladding, excellent irradiation stability, low reactivity with water, and low fabrication costs compared with metallic fuels, carbides, or nitrides. Pellets are generally preferred although considerable attention has been given to vibratorily compacted fuel, particularly for recycle. Because of service requirements, tight specifications must be met on trace element levels and fuel density; and for pelletized fuels, the shape of the pellet and the presence of chips or laminations are important. The irradiation behavior of UO_2 is influenced markedly by its low thermal conductivity and vaporization behavior. Under normal operating conditions, the fuel may be divided into three temperature zones, each of which has its individual character regarding fission-gas release and fuel swelling. At heat ratings in current reactors both thermal stress cracking and void redistribution occur, affecting performance. With proper design, peak heat ratings of about 20 kW/ft to burnups of 30,000 MWd/mt are permissible. Transients producing gross center melting can be tolerated provided there is sufficient free volume to accomodate the 7% fuel volume increase and the fission product gases from the melted zone and provided the subsequent thermal history is controlled. (auth)

NSA, vol. 23, 1969

4904 SOME PROBLEMS IN MANUFACTURING FACTORY OF NUCLEAR FUELS. Seki, Y. (Mitsubishi Nuclear Fuel Co., Tokai, Ibaraki (Japan). Tokai Works). Genshiryoku Kogyo; 20: No. 6, 35-39(Jun 1974). (In Japanese)

The features are described of the industry making uranium fuel for light water reactors. Conversion of UF_6 to UO_2 powder, forming of UO_2 pellets and construction of fuel rods are standard processes. However, handling must be carried out carefully. Uranium is under international control, is radioactive, has a critical limit quantity, and is expensive. UF_6 gas is poisonous and corrosive. Various considerations are necessary for fuel rods because the behavior of the fuel rods in operating reactors is not clearly known. Abnormal behavior reported includes local damage of cladding due to hydriding, compacting due to the heating of UO_2 pellets, collapse of claddings, and bending of fuel rods. Other problems are the control of dimensions of fuel rods, management of enrichment, inspection of damage or fault of claddings and pellets, management of uranium loss in the fabrication process, programs for recycle of uranium, and methods of reliable inspection. (JA)

NSA, vol. 33, 1976

Fabrication of Vibrasol Fuel Rods, *P. F. Sens (Interfuel B.V.—The Netherlands), J. B. W. Kanij, K. A. Nater (KEMA—The Netherlands), J. H. N. Verheugen (RCN—The Netherlands)*

INTRODUCTION

Due to the presence of an established pellet technology, a breakthrough of vibratory compaction as an alternative fuel rod production method may be expected if distinct advantages in fabrication and operating behavior of vibrocompacted fuel can be obtained. Along these lines, development work with positive results has been performed in the Netherlands since 1968 and, at present, one of the main objects of the Interfuel Company, founded in 1972, is to enlarge the scale of the existing laboratory experience through pilot plant operation to a full-size industrial activity. The fuel material for vibrocompaction consists of spherical particles, produced with a sol-gel-type process. Proper size fractions of this fuel can be compacted to high density by vibration and for this reason the complete process has been called the vibrasol process. The following merits of the vibrasol process as compared to other vipac processes may be mentioned:

1. direct preparation of the various size fractions with a narrow size distribution, instead of obtaining fractions by grinding or milling of larger particles
2. no dust hazard
3. extensive remote control possibilities, of special importance when working with mixed oxides
4. improved density distribution in the fuel rods
5. improved irradiation performance.

This paper describes recent developments in the vibrasol process. Another paper at this conference[1] gives the results of irradiation experiments and performance tests of vibrasol fuel rods and elements.

PRODUCTION OF SPHERICAL FUEL PARTICLES

Principles

To reach high densities in vibratory compacted fuel, the density of the individual particles must be as high as possible. Such particles can be obtained by crushing larger lumps of fused or electro-deposited oxide. This method produces a wide size distribution with a low efficiency, causes dust problems and is liable to an uncontrolled moisture pickup from the environment. A process by which the particles are made directly in the correct size with a narrow size distribution has to be preferred. Various so-called sol-gel processes exist, in which the individual particles are formed by solidification of droplets in suspension in a liquid. The process applied by Interfuel has many aspects in common with the sol-gel processes, but actually does not imply the preparation of a sol, from which the droplets are formed. Interfuel has exclusive rights on the KEMA U(VI)-process,[2,3] involving the internal gelation of droplets of a substoichiometric uranyl-nitrate solution, mixed with a solution of hexamethylenetetramine and urea, to give spheres of ammonium diuranate. The gelation occurs by contacting the droplets with a hot, immiscible organic liquid. To produce mixed-oxide particles, plutonium can be added to the uranyl nitrate solution.

In general, three particle size fractions in the range of 1, 0.1, and <0.04 mm, as-sintered, have to be used to reach vipac densities equivalent to water reactor fuel. For lower density fast reactor fuel, two fractions may be sufficient. For the coarse fraction the droplets are generated by free formation on the capillary end of hypodermic needles. This technique is well established, has sufficient production capacity by increasing the number of needles, and produces particles in a narrow size distribution: average diameter after sintering of 1000 μm with a standard deviation of 70 μm. To disperse the medium fraction, with an average sintered diameter of 1000 μm, a spray nozzle technique was used originally, producing a wide size distribution with standard deviation of 60 μm and a correspondingly low efficiency of about 45% material within the required size limits [Fig. 1(a)]. To improve this, a new technique has been developed with a standard deviation of about 15 μm and an efficiency increase to about 90% [Fig. 1(b)]. In this technique, jets of the fuel-containing solution are chopped by interaction with a fast flowing film of organic medium to form droplets of a discrete size distribution under the influence of the surface tension.

To produce the fine fraction, the fuel-containing solution is dispersed as an emulsion in a carrier liquid, then the temperature is raised to cause internal gelation and the liquid is boiled off by further temperature increase. This method produces spherical particles of about 10-μm diam after sintering. After gelation, the particles of all fractions are separated from the gelling liquid and washed with diluted ammonia in a way that is determined by the requirements of each fraction separately. Consequently, the particles are dried in air, calcined, and sintered in a reducing atmosphere to above 99% TD.

Pilot Plant Facilities

Based on the above process, two pilot plant facilities have been constructed and put in operation in 1974. The first pilot plant produces coarse, medium, and fine fraction of UO_2 with a maximum enrichment of 4.5% ^{235}U. The amount of fissile material is controlled by limitation of the batch size, except in the geometrically safe storage vessels. The plant consists of a dissolution section to provide the uranyl nitrate solution; a storage section; a mixing vessel for uranyl nitrate and hexa-urea solution; three separate systems for the dispersion and gelation of coarse, medium, and fine fraction; and separation, washing, and drying equipment.

The purpose of the pilot plant is to:

1. further develop a number of process steps and scale up to production size
2. prepare material in sufficient quantities for vibratory compaction development
3. produce fuel material for experimental irradiations and performance tests.

The pilot plant has been built as a modular system, allowing easy modifications. With a semicontinuous op-

(Cont'd pg. 28)

eration, the yearly capacity of the pilot plant is about 6 MT. In addition, a smaller pilot plant has been constructed in glove boxes to produce only the coarse fraction, but in mixed $(U, Pu)O_2$. Except for the addition of the Pu, the process is the same as for UO_2.

VIBROCOMPACTION

Important objectives for compaction are (a) small deviations in smeared fuel density in the fuel rods, (b) small density variations in the axial direction of a fuel rod, and (c) obtaining a variety of densities according to specification requirements.

In the initial phase of the vibrasol development, many fuel rods have been made with a spread in smeared densities up to 3% and an axial density variation up to 5%, while the maximum smeared density was limited to about 88% TD. This was caused by the wide spread in particle properties with respect to density and size distribution of the fuel materials produced in this period. Compaction experiments with the fuel material, produced by the improved methods described above, show that the spread in the smeared fuel density and in the axial density variations can be decreased to 2%, when high-density fuel particles above 99% TD and well-defined particle fractions are used. Furthermore, by proper selection of the size fractions, overall densities up to 92% TD are obtained. The compaction is carried out with an electrodynamic shaker. Preferably, the fuel rod is clamped on a horizontal resonance beam and attached to the shaker. In this way high accelerations are induced in the fuel rod in both the horizontal and vertical directions. Compaction is obtained by continuous sweeping through the resonance frequencies between 400 and 3000 Hz.

FUEL ROD FABRICATION

Many features of the fuel rod fabrication can be related to the already well established fabrication procedures used for fuel rods with pelletized fuel. Special attention has to be paid to obtain low hydrogen contamination of the fuel. A low hydrogen content of < 1 ppm, including absorbed gases and moisture, can be obtained with special drying procedures, particularly for the fine fraction of about 10 μm, and by working in a well-controlled dry atmosphere during vibrocompaction (dew-point -60°C). Fuel rods have been made for irradiation experiments in the high flux reactor at Petten and in the Halden boiling water reactor under PWR and BWR conditions, respectively. Also, two prototype fuel elements have been fabricated. Irradiation in the reactor at Dodewaard, which started in 1972, is still being continued. Recently a number of fuel rods with mixed oxide have been fabricated and irradiation of these rods in the HFR was started in September 1974.

1. P. F. SENS and E. B. M. MAJOOR, "Irradiation Behaviour of Vibrasol Fuel," First European Nucl. Conf., Paris (Apr. 1975).

2. F. W. HAMBURG et al., "Development of Vibrocompacted Fuel Elements with UO_2 Fuel Prepared by the Sol-Gel Process and by Electro-Deposition," *Proc. 4th Intern. Conf. Peaceful Uses At. Energy, Geneva*, Vol. 8, p. 263, Vienna, UN and IAEA (1972).

3. J. B. W. KANIJ et al., "The KEMA U(VI)-Process for the Production of UO_2 Microspheres," *Proc. Panel Sol-Gel Processes for Fuel Fabrication*, Vienna, IAEA-161, p. 185 (May 1973).

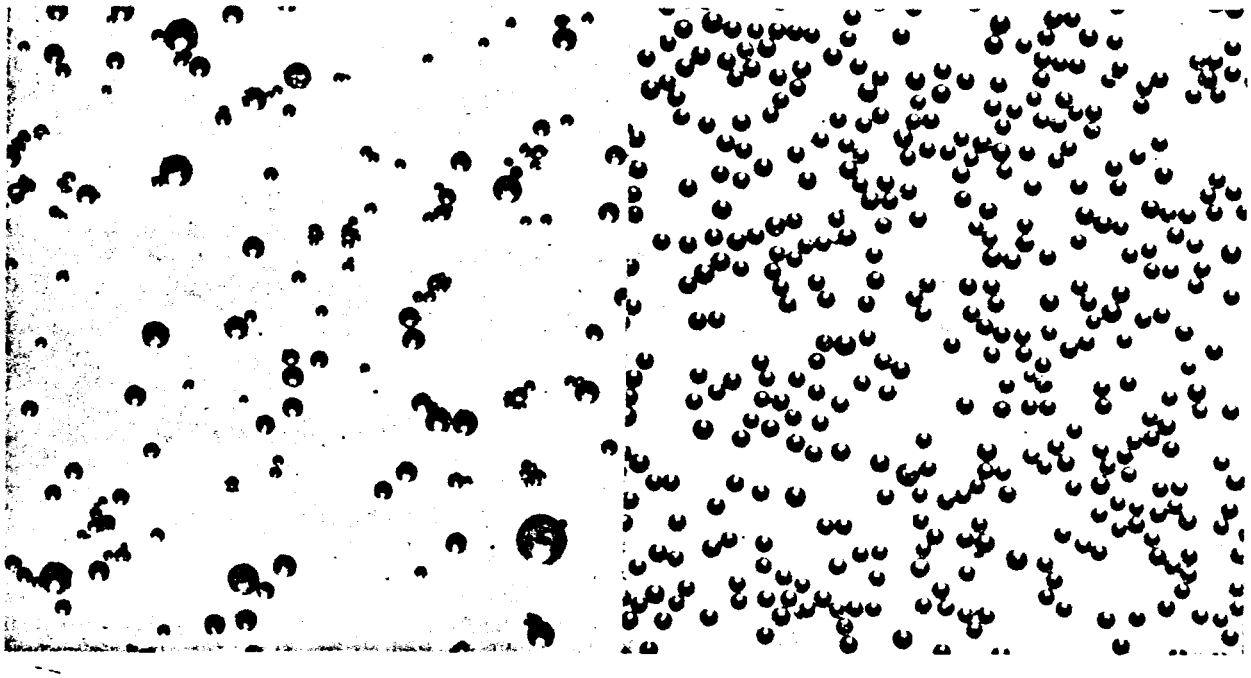


Fig. 1. Medium fraction vibrasol fuel, produced with the old (a) and new (b) method.

2588 CONTRIBUTION TO THE ENERGETIC STUDY OF COLD PRESSING OF UO_2 POWDERS. Stamenkovic, I. D.; Zivanovic, B. M.; Ristic, M. M. (Boris Kidric Inst. of Nuclear Sciences, Belgrade. Univ. of Nis, Yugoslavia). Phys. Sintering; 3: No. 3, 157-65 (Sep 1971).

Results of pressing experiments using a double-acting hydraulic press to compact UO_2 powders are described. It was found that the behavior of uranium dioxide powders during pressing is directly related to their physical properties. The total work of pressing was higher for powders having a larger surface area, as the result of their contraction. The pressing work dissipates on agglomerate fragmentation and the friction between particles amounts up to 30% of the total pressing work, mainly in the pressure range up to 3 ton cm^{-2}. In this pressure range the increase of compact surface area is the greatest. (J.R.D.)

NSA, vol. 26, 1972

20846 (AED-CONF-71-100-26) PROGRESS IN ZIRCALOY-4 CANNING TECHNOLOGY FOR PWR FUEL. Stehle, H. (Siemens A.G., Erlangen (West Germany)). May 1971. 21p. (CONF-710901-524). Dep. NTIS (U. S. Sales Only).

From fourth international conference on the peaceful uses of atomic energy; Geneva, Switzerland (6 Sep 1971).

Progress is reported in Zircaloy-4 canning technology for PWR fuel in the following areas: investigation of biaxial mechanical properties (bursting, creep); improvement and simplification of surface treatment; and welding in pressurized atmosphere. (D.H.M.)

NSA, vol. 26, 1972

8028 (BAW-10097(Rev.1)) MARK C (17 x 17) FUEL ASSEMBLY. Research and Development. Revision 1. Stoudt, R. H.; Kern, R. C.; Walton, L. A. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Nuclear Power Generation Div.). Apr 1975. 14p. Babcock and Wilcox, Lynchburg, VA.

The research and development programs related to the B and W Mark C (17 x 17) fuel assembly are described. The programs described herein are designed to provide data for analytical models, confirm predictions, and verify the adequacy of the design. A general description of the Mark C assembly is presented along with comparisons to the Mark B (15 x 15) fuel assembly. The report is organized in three main test areas: mechanical and hydraulic; critical heat flux; and rod swelling and burst. The test program scopes and objectives are discussed along with the test methods. (auth)

NSA, vol. 32, 1975

10382 NUCLEAR FUEL ELEMENT. Takeda, K.; Hirose, Y. (to Hitachi Ltd.). Japanese Patent 1973,11274. 18 May 1966. 3p. (In Japanese).

A nuclear fuel element, which is densely filled with powders of different production methods is proposed with a view to reducing the amount of impurities which might otherwise be contained in the fuel element and which are volatile at high temperature. The powders for the fuel element are composed of nuclear fuel powders having a plurality of particle sizes, with the coarser and intermediate powders made of fused nuclear fuel material having high density and the finer powders made of sintered nuclear fuel material. (JA)

NSA, vol. 31, 1975

2593 CURRENT STATE OF PRODUCTION TECHNIQUES FOR NUCLEAR FUELS IN JAPAN. Taketani, Kiyoaki (Japan Atomic Energy Research Inst., Tokai). Karyoku Hatsuden; 22: No. 3, 257-64(Mar 1971). (In Japanese).

The status of production methods for nuclear fuels for light-water reactors (primarily BWR and PWR) in Japan is discussed. Production processes and facilities for the purpose are satisfactory, and the quality control and inspection techniques are at a high level. Products also showed high performance in demonstration tests. Techniques for producing nuclear fuels includes conversion of UF_6 to UO_2 for light-water reactor use which predominates in this field. The fuel being fabricated is mainly UO_2 pellets enclosed in Zircaloy tubes. Fuel inspection methods are being studied at present. Demonstration tests of fuels for JPDR are described. By participating in the Halden Project, the development progresses steadily. As for test fuel assemblies, test results as well as their structure are described. (Japan)

NSA, vol. 26, 1972

59183 FUEL FABRICATION. Thomas, Ivor (Kerr-McGee Corp., Oklahoma City). pp 125-35 of Education and Research in the Nuclear Fuel Cycle. /Elliott, David M. (ed.). Norman, Okla.; Univ. of Oklahoma Press (1972).

From Meeting on education and research in the nuclear fuel cycle; Norman, Okla. (5 Oct 1970). See CONF-701035.

The fabrication of nuclear reactor fuel elements for use in the commercial production of electric power is discussed. For the fabrication of fuels consideration must be given to the environment it must withstand, design parameters that must be considered in order to achieve the desired performance, fuel-cycle cost economics, and plant capacity effects. A typical manufacturing process for the production of water reactor fuels is given. It was concluded that there is some overcapacity in the UO_2 fuel fabrication industry and serious overcapacity in the mixed oxide fuel industry through 1974, rapid growth in capacity beyond 1975 will be a necessity. (P.C.H.)

NSA, vol. 26, 1972

263417 Thompson, J.R.; Rowland, T.C. General Electric Co., Schenectady, N.Y. (USA). **Nuclear fuel element.** (In German). Kernbrennelement. **German (F.R.) patent document 2501309/A/.** Int. Cl. G21C 3-20. 14 Aug 1975. 23 p. 8 figs.

The fuel element for, e.g., a PWR or BWR contains an intermediate layer, e.g. a cylinder or a foil, between the fuel and the fuel can. This intermediate layer inhibits the mechanical interaction between fuel can and fuel, insulates the generated fission products from the fuel can and improves the axial thermal gradient along the fuel element. The intermediate layer or lining consists of a refractory metal such as, e.g., molybdenum, tungsten, rhenium, niobium, or their alloys. The fuel can consists of, e.g., zirconium, zirconium alloys, stainless steel, aluminium or aluminium alloys. With this combination of materials for fuel can and lining, the fuel may consist of uranium compounds, plutonium compounds, or thorium compounds. (DG/AK).

FUEL ELEMENTS: protective coatings.

Atomindex, vol. 7, 1976

3,909,370
PROCESS FOR SURFACE TREATMENT OF ZIRCONIUM-CONTAINING CLADDING MATERIALS FOR FUEL ELEMENTS OR OTHER COMPONENTS FOR NUCLEAR REACTORS

Ketil G. Videm; Liv R. Lunde, and Henk H. Kooyman, all of Kjeller, Norway, assignors to Institutt for Atomenergi, Kjeller, Norway

Continuation of Ser. No. 52,759, July 6, 1970. This application Nov. 30, 1972, Ser. No. 310,787

Int. Cl. C25d 5/34

U.S. Cl. 204—32 R 5 Claims

1. A method for surface treatment of zirconium to produce a protective coating thereon, said method consisting of:

cleaning said zirconium by submersing the same in a pickling bath containing fluoride ions;

removing said zirconium from the pickling bath and then while still contaminated with fluoride containing residues, subjecting the same to anodic oxidation in the absence of additional fluoride ions until said fluoride containing residues are rendered incapable of interfering with the subsequent oxidation of said zirconium to produce a protective oxide coating thereon; and

thereafter producing a protective oxide coating on the clean zirconium by subjecting the same to surface oxidation conditions.

U.S. PATENTS 1975

35985 (ANL-7917) DEVELOPMENT STUDIES ON A FLUIDIZED-BED PROCESS FOR CONVERSION OF U/Pu NITRATES TO OXIDES. PART 1. LABORATORY-SCALE DENITRATION STUDIES. Vogler, S.; Grosvenor, D. E.; Levitz, N. M.; Teats, F. G. (Argonne National Lab., Ill.). Apr 1972. Contract W-31-109-eng-38. 31p. Dep. NTIS.

Laboratory experiments were carried out to simulate the fluid-bed denitration of uranyl nitrate–plutonium nitrate solutions. These experiments indicated the denitration product to be UO_3–PuO_2, which yielded UO_2–PuO_2 upon hydrogen reduction. From this UO_2–PuO_2 product, pellets of 89% theoretical density were prepared by sintering in argon at 1600°C. Electron microprobe examination of the pellets indicated good homogeneity with no evidence of isolated particles of plutonium oxide. The cosolubility of uranyl nitrate and plutonium nitrate (1-2M U + Pu) in nitric acid was measured. The invariant point was not reached for solutions containing 0.67 fraction plutonium (Pu/U + Pu). (auth)

NSA, vol. 26, 1972

4908 NUCLEAR FUEL ROD. Wakasugi, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1974-32911/B. 11 Jun 1970. 4p. (In Japanese)

In a cladding tube charged with fuel pellets and sealed at both ends, a fuel rod is disclosed in which the so-called ridging or bambooing of the cladding tube is prevented by employing barrel-shaped fuel pellets with bulging sides. Pellets provided with a suitable degree of bulge will swell and deform during irradiation but will assume near cylindrical proportions as a result of their unique barrel shape. The cladding tube is therefore not unduly stressed which accordingly permits continuous operation even at high thermal output without fear of ridging and eventual rupture. (JA)

NSA, vol. 33, 1976

20728 MODEL STUDIES ON CONICAL FLUIDIZED BEDS FOR COATING OF NUCLEAR FUEL PARTICLES. Wallroth, C. F.; Gyarmati, E.; Nickel, H. (Kernforschungsanlage, Juelich, Ger.). Chem.-Ing.-Tech.; 43: No. 24, 1298-1304(Dec 1971). (In German).

The pyrolytic coating of nuclear fuel particles is carried out in conical fluidized beds at temperatures up to and above 2000°C, thus precluding direct observation of movements in the bed. Principles involved in performing model studies at room temperature for simulation of the movements are reported, as are the experimental results for a solid/gas system (petunia seeds/hydrogen) using a double-channel jet system. Varying distribution of gas between the central and the annular jets permits establishment of various forms of movement. The best results with regard to uniformity of coating were obtained in coating experiments with unperturbed bubble movement. (auth)

NSA, vol. 26, 1972

32060 RECOVERY OF URANIUM OXIDE SCRAP. (to Westinghouse Electric Corp.). British Patent 1,228,525. 15 Apr 1971. Priority date 4 Apr 1968, United States.

A method is described for recovery of uranium in one step from uranium oxide scrap originating from the production lines for fuel pellets and containing impurities such as Si, B, and C derived from the grinding wheel and/or the grinding lubricant. The U scrap and the impurities are reacted with elemental fluorine at a temperature ranging from 300 to 500°C to form a product containing UF_6 and the volatile fluorides of the impurities. All solids are removed from the gaseous products, and the gaseous products are then cooled to a temperature below the boiling point of UF_6 and above the boiling point of the volatile fluorides of the impurities so that only the UF_6 condenses and is recovered free from impurities. (B.L.M.)

NSA, vol. 25, 1971

19718 (WCAP-4167-2, Sect.4) DEMONSTRATION ASSEMBLY DESIGN. (Westinghouse Electric Corp., Pittsburgh, Pa. Nuclear Fuel Div.). Mar 1971. 43p.

In EEI-Westinghouse plutonium recycle demonstration program progress report for the period ending December 1970.

Four fuel assemblies containing 720 PuO_2–UO_2, Zircaloy-4 clad fuel rods were inserted in the San Onofre power reactor during the first refueling. These plutonium recycle demonstration assemblies will be operated through the normal three reactor cycles, with removable rods subjected to interim examinations. The nuclear design, mechanical design, thermal-hydraulic design, and fuel pellet design of the recycle assemblies are summarized. Safety considerations are analyzed for a control rod ejection accident and for fuel handling. (H.D.R.)

NSA, vol. 26, 1972

19719 (WCAP-4167-2, Sect.5) DEMONSTRATION ASSEMBLY FABRICATION. (Westinghouse Electric Corp., Pittsburgh, Pa. Nuclear Fuel Div.). Mar 1971. 78p.

In EEI-Westinghouse plutonium recycle demonstration program progress report for the period ending December 1970.

A portion of the plutonium recycle demonstration program required the fabrication of 734 mixed-oxide fuel rods and 16 UO_2 replacement fuel rods. The fabrication facility is described, and all operations are summarized, including: receipt of plutonium nitrate solution; conversion to PuO_2 or mixed-oxide; blending the mixed-oxide; pellet fabrication; rod fabrication and inspection; recovery, purification and recycle of process scrap; and treatment of waste solids and solutions for disposal. (H.D.R.)

NSA, vol. 26, 1972

23289 CURRENT BWR FUEL DESIGN AND EXPERIENCE. Williamson, H. E.; Ditmore, Dana C. (General Electric Co., San Jose, Calif.). Reactor Technol.; 14: 68-98(Spring 1971).

An up-to-date review of experience and development data on boiling-water-reactor (BWR) Zircaloy-clad UO_2 fuel is presented. The area of technology demonstrated by irradiations of Zircaloy-clad UO_2 fuel rods and pins in reactors, loops, and capsules goes beyond the combination of fuel-rod power and exposure that will be experienced in modern BWRs not only for normal continuous operation but also for anticipated power transients. The recorded successful irradiations demonstrate that not only safe but reliable fuel can be designed for modern BWR conditions. A value of 1% plastic strain of Zircaloy cladding is defined as the limit below which fuel damage due to overstraining is not expected to occur during any single power transient. The linear heat-generation rate required to cause this amount of cladding strain is ~28 kW/ft in fresh fuel. The analytical and experimental bases employed in the justification of this limit are reviewed. 32 references. (auth)

NSA, vol. 25, 1971

25972 EXPERIENCE WITH BWR FUEL. Williamson, H. E.; Ditmore, D. C. Technical Meeting No. 5/5. Basel; Nuclex (1972). 10p.

From international nuclear industries fair; Basel, Switzerland (16 Oct 1972).

A review of experience with BWR Zircaloy-clad UO_2 pellet fuel is presented. Experience on the design, manufacture, quality control, and operation of a large volume of Zircaloy-clad UO_2 pellet fuel over the past 10 to 12 years has provided favorable feedback on the adequacy of the design for a commercial power reactor environment. Design and manufacturing improvements to minimize the incidence and effect of material variabilities, flaws, and impurities which occur statistically in large volume production processes are discussed. 6 references. (auth)

NSA, vol. 29, 1974

256084 Wood, J.C. **Nuclear fuel element and a method of manufacture thereof. US Patent 3,901,761.** Int. Cl.G21c3/20. 26 Aug 1975. 6 p.

A nuclear fuel element having a sheath of zirconium or a zirconium alloy and a cross-linked siloxane lacquer coating on the inner surface of the sheath and separating the nuclear fuel material from the sheath is described. The siloxane lacquer coating retards cracking of the sheath by iodine vapor emitted by the fuel during burn-up, and acts as a lubricant for the fuel to prevent rupture of the sheath by thermal ratchetting of the fuel against the sheath and caused by differential thermal expansion between the fuel and the sheath. A silicone grease is applied as a thin layer in the sheath and then baked so that oxidative cleavage of the side chains of the grease occurs to form a cross-linked siloxane lacquer coating bonded to the sheath.
FUEL CANS: coatings.

Atomindex, vol. 7, 1976

7216 URANIUM DIOXIDE ARTICLE AND METHOD OF PRODUCING THE SAME. Wyatt, Brian S. (to Atomic Energy of Canada Ltd.). U. S. Patent 3,609,095. 28 Sep 1971. Filed 24 Feb 1969.

A method is described for the production of uranium dioxide articles such as nuclear fuel elements and ceramic shapes and which comprise a powdered mixture of UO_{2+N} and U or Pu metal, wherein X is the amount of excess O_2 required to produce, from said mixture, stoichiometric UO_2 when subjected to reaction conditions under irradiation. One of the problems incidental to the use of nuclear fuel elements containing vibration packed mixtures of uranium dioxide (UO_2) powders is the shrinkage of the UO_2 that occurs at the high temperatures generated in the fuel element during irradiation. The maximum packed densities thus far obtained in nuclear fuel elements, made by vibration packing inside a metal sheath a mixture of different size fractions of UO_2 powders, are in the range of 88 to 92 percent of the theoretical density. The UO_2 powder is commonly prepared by crushing previously sintered or fused blocks of UO_2. During irradiation of such an element, the UO_2 powder tends to sinter with resultant shrinkage of the UO_2 body. This shrinkage could impair heat transfer between fuel and sheath and might eventually result in the collapse of the fuel-sheathing when this is insufficiently thick to be self-supporting in the reactor coolant pressure. (Official Gazette)

NSA, vol. 26, 1972

Blank Page

II. EFFECT OF FUEL IMPURITIES ON FUEL PERFORMANCE AND CLADDING ATTACK (Both SS304 and Zircaloy 2 and 4)

Blank Page

25961 MECHANICAL INTERACTION BETWEEN FUEL AND CLADDING. Aas, S. (Institutt for Atomenergi, Kjeller, Norway). pp 165-200 of Advanced Course on Limiting Aspects of Fuel Element Performance in Water Cooled Power Reactors. Kjeller, Norway; Institutt for Atomenergi (1971).

Mechanical interaction between UO_2 pellet fuel and the cladding is due to greater expansion of the fuel because of its higher temperature and to fuel swelling and cracking. The extent and nature of the interaction is dependent on the gap, length of the pellets, end shape of the pellets, heat load, rate of power increase, and cladding properties. The stress-strain pattern usually has a changing multiaxiality with consequent varying strains to fracture, and the strains may be concentrated over small lengths of the cladding. The properties of Zircaloy are adversely affected by irradiation and to avoid difficulties the pellet shape, dimensions and density, and cladding properties should be chosen with due regard to reactor system conditions. The way the reactor is loaded and operated is also a factor determining fuel life. Many of the processes described are not known precisely and there is need for many more in-pile measurements. In addition the measurement of heat production and channel power should be to a similar degree of accuracy. (Norway)

NSA, vol. 26, 1972

42525 STRAIN ON FUEL ROD CANS IN PRESSURIZED WATER REACTORS. Aisch, F. W.; Fuchs, H. P.; Holzer, R. (Siemens AG, Erlangen, Ger.). pp 554-7 of Reaktortagung, Bonn, 1971. Bonn; Deutsches Atomforum E. V. (1971). (In German).

From Reactor meeting; Bonn, Germany (30 Mar 1971). See CONF-710346.

The fuel rod power of rods exposed to different operating conditions is considered. The mechanical strain on the fuel cans and their strength criteria are discussed. (INIS)

NSA, vol. 26, 1972

2. Fuel-Element Performance in Pressurized Water Reactors / *R. J. Allio (W-APD) Invited Paper*

Westinghouse is utilizing both stainless steel and Zircaloy as cladding for oxide fuel elements in pressurized water reactors. Operating experience to date is summarized in Table I. Reliability of fuel elements clad with both materials has been excellent.

Fuel elements clad with annealed Type-348 stainless steel in Yankee and cold-worked Type-304 stainless steel at Saxton continue to accumulate higher burnup. Post-irradiation evaluation of sample fuel elements confirms the similarity in behavior of annealed and cold-worked stainless steel in pressurized water environments. Mechanical property changes in annealed material approach saturation after an integrated exposure (> 1 MeV) of 3×10^{21} n/cm^2. Additional information from post-irradiation examination of stainless-steel-clad fuel has been reported previously[1,2].

To reduce fuel-cycle costs, future cores will probably be fueled with Zircaloy-clad elements, although the use of thin-walled collapsed stainless steel remains a possibility. Zircaloy in CVTR and Saxton has performed as predicted from out-of-pile experiments. Corrosion has been identical in the borated and unborated Saxton Reactor environment. Tests of Zircaloy in Saxton have also demonstrated that corrosion is independent of surface treatment prior to irradiation. Zircaloy-clad fuel elements deliberately defected either prior to or after irradiation have operated satisfactorily during continued irradiation.

Because of its lower cost and higher reliability, pelletized fuel is reference material for Westinghouse reactors

TABLE I

Irradiation History for Westinghouse Fuel Elements

(July 15, 1965)

Reactor	Cladding Material	Cladding Thickness, In.	Peak Power Density kW/ft	Peak Exposure	
				n cm^{-2} > 1 MeV ($\times 10^{21}$)	MWd/t of uranium
Yankee	Annealed 348 Stainless	0.021	11.7	8	38 000[a]
[illegible]-3	Annealed 348 Stainless	0.0205	11.6	2	9 000
[illegible]	Cold-Worked 304 Stainless	0.015	13.5	3	22 500[a]
[illegible]	Cold-Worked 304 Stainless	0.010[b]	13.5	1	10 000
[illegible]	Cold-Worked 304 Stainless	0.015	10.5	1	6 000[a]
CVTR	Zircaloy-4	0.023	14.0	8	14 000[a]
Saxton	Zircaloy-4	0.023	16.0	2	14 000

[a] Irradiation Continuing
[b] Collapsed Cladding

fueled with UO^2. However, both pelletized and vibratory-compacted fuel are undergoing irradiation as part of the Saxton Plutonium Program. Mixed-oxide (PuO_2-UO_2) assemblies will experience an estimated peak burnup of 20 to 30 000 MWd/t of uranium at 16 kW/ft in the Saxton Reactor. The experimental program will permit direct comparison of the effects on performance of fuel composition (UO_2 versus UO_2-PuO_2), fuel-manufacturing process (pelletizing versus vibratory compaction) and cladding (Zircaloy versus stainless steel). Characteristics of the Zircaloy-clad plutonium fuel (UO_2-6.6 wt% PuO_2) assemblies are summarized in Table II.

ANS TRANS., vol. 8, 1965

TABLE II

Zircaloy Assemblies - Saxton Plutonium Program

	Pelletized	Vibratory Compacted
Number of Assemblies	7	2
Rods Per Assembly	72	72
Cladding Thickness, In.	0.023	0.023
Cladding Outside Diameter, In.	0.391	0.391
Fuel Density, % of Theoretical	94	87
Diametral Gap, In.	0.005	0

1. SMALLEY, W. R., "Evaluation of Highly-Irradiated Yankee Fuel Cladding," *Trans. Am. Nucl. Soc.*, 7, 2, (1964).

2. McCLINTOCK, D. R., et al., "Performance of Thin-Walled Stainless Steel Clad Rod Type Fuel Elements in Saxton." *Trans. Am. Nucl. Soc.*, 7, 2, (1964).

ANS TRANS., vol. 8, 1965

28548 (WARD-4210-T-3-8) OXIDE FUEL ELEMENT DEVELOPMENT QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING JUNE 30, 1971. Anderson, C. A, Jr.; Bishop, E. C.; Boltax, A.; Ray, W. E. (Westinghouse Electric Corp., Madison, Pa. Advanced Reactors Div.). Contract AT(30-1)-4210. 150p. Dep. NTIS.

Research on fuels is reported related to radiation effects on fuel-cladding interactions and fuel swelling. Results are also included on oxide fuel pin irradiation tests, irradiation tests on grid-type subassemblies, design and fabrication of grid-type subassemblies, fabrication of PuO_2–UO_2 pellets, and fuel evaluation programs. (J.R.D.)

NSA, vol. 26, 1972

27599 EFFECTS OF RADIATION ON THE OXIDATION AND HYDROGEN ABSORPTION OF ZIRCONIUM ALLOYS IN STEAM. Asher, R. C.; Davies, D.; Hall, A.; Kirstein, T. B. A.; Marriott, J. W.; White, P. J. (Atomic Energy Research Establishment, Harwell, Eng.). Electrochem. Technol., 4: 231-6(May-June 1966).

The oxidation of Zircaloy-2 and of Zr–2½% Nb alloy in steam at 300 to 340°C is enhanced by an average of eightfold under irradiation in a reactor at $\sim 3 \times 10^{13}$ n.cm^{-2} sec^{-1} (fast) + $\sim 5 \times 10^{8}$ R.hr^{-1} γ. The magnitude of the effect depends on the temperature, the radiation intensity, and whether pretransition or posttransition oxidation is involved. Limited evidence suggests that there is no effect on the proportion of hydrogen absorbed by the metal. It is concluded that the enhancement of oxidation is brought about by fast neutron damage in the oxide film, or possibly in the metal substrate. This view is supported by: experiments under ionizing radiation, in the absence of neutrons; observations on the effects of adding small amounts of hydrogen or oxygen to the steam; and a study of the postirradiation oxidation. A tentative mechanism is discussed. (auth)

NSA, vol. 20, 1966

36624 EFFECTS OF RADIATION ON THE CORROSION OF SOME ZIRCONIUM ALLOYS. Asher, R. C.; Davies, D.; Kirstein, T. B. A.; McCullen, P. A. J.; White, J. F. (Atomic Energy Research Establishment, Harwell, Eng.). pp 178-85 of Proceedings of the Fourth International Congress on Metallic Corrosion. /Hamner, Norman E. (ed.). Houston, Tex.; National Association of Corrosion Engineers (1972).

From fourth International Metals Corrosion Congress proceedings; Amsterdam, Netherlands (7 Sep 1969).

Studies were made on the effects of reactor radiation on the corrosion of Zircaloy-2, Zr–2.5% Nb, and Zircaloy-4. Intensities of neutrons and of γ radiation typical of a power reactor, viz $\sim 4 \times 10^{13}$ n cm^{-2} sec^{-1} fast neutrons (>1 MeV), $\sim 4 \times 10^{14}$ n cm^{-2} sec^{-1} thermal neutrons, and $\sim 10^{9}$ R h^{-1} γ. The results showed that irradiation can enhance the corrosion of Zircaloy-2 and of Zr–2.5% Nb alloy by factors of the order of 10. The enhancement was most significant in the post-transition region and in the temperature range 250 to 350°C; it disappeared at 400°C and above. With Zr–2.5% Nb alloy, a slight suppressing effect of radiation in the early part of the pre-transition region was observed under certain conditions. Detailed corrosion behavior under radiation was quite different in gaseous corrodents (steam and impure CO_2) from that in liquid water. A consistent effect of irradiation on the percentage hydrogen pick-up during corrosion was not established. (P.C.H.)

NSA, vol. 26, 1972

40429 PLUTONIUM AND FISSION PRODUCT REDISTRIBUTION IN MIXED-OXIDE FUELS DURING IRRADIATION. Bahl, J. K. (Bhabha Atomic Research Center, Bombay); Freshley, M. D. Nucl. Technol.; 15: No. 4, 114-24(Aug 1972).

Plutonium and fission product migration were studied in a low burnup vibrationally compacted UO_2–2 wt% PuO_2 specimen utilizing autoradiography, microsampling, microgamma scanning, and shielded electron microprobe analytical techniques. The results show that a significant change in radial plutonium concentration can occur rapidly in high performance mixed-oxide fuels by a vapor transport mechanism that is coincident with the fuel restructuring phenomenon. This results from the condensation of uranium-rich vapor in the cooler peripheral regions of the fuel rod. Cesium, ruthenium, and rhodium migrate significantly in mixed-oxide fuels, whereas zirconium, niobium, cerium, and praseodymium do not redistribute significantly. (auth)

NSA, vol. 26, 1972

30223 BEHAVIOR OF FUEL ROD CLADDING USED IN LIGHT-WATER REACTORS. Tokyo; Nuclear Safety Research Association (1970). 118p. (In Japanese).

The results of studies on the behavior of fuel cladding rods in cores of light-water reactors at normal operation and accidents are presented: mechanical properties, anti-internal pressure resistance in particular, of Zircaloy cladding material; the behavior of fuel cladding rods at the time of a loss-of-coolant; fuel cladding rods and their integrity; the behavior of hydrides in cladding materials and the effects; and the reactions between zirconium and coolant water. (Japan)

NSA, vol. 26, 1972

1. The Behavior of Fission-Product Gases in Uranium and Uranium Dioxide*, *R. S. Barnes, R. G. Bellamy, B. T. Bradbury, J. B. Sayers, and A. D. Whapham (UKAEA-Harwell).*

The fission products xenon and krypton present problems in both uranium and uranium dioxide fuel elements. In the metal, gas bubbles form and can cause an appreciable volume increase or "swelling." In the oxide, current evidence suggests that, although bubbles form[1], swelling is not normally a problem in conditions of interest in gas-cooled reactors; however, a significant proportion of the gas is released[2]. In both cases the fuel cladding may be stressed, with consequent deterioration of the heat transfer and eventual rupture of the cladding. It is believed that appreciable swelling of uranium and the appreciable release of gas from uranium oxide both result, not from the migration of single inert-gas atoms, but from the bodily migration of the bubbles of fission gas. Such migration (and low solubility) has recently been demonstrated during direct observation of both copper[3] and uranium dioxide[4] in the electron microscope.

In uranium the tendency to swell has been reduced by introducing precipitates that are believed to anchor the bubbles, which are thus prevented from migrating so that they do not coalesce and increase their total volume due to the fall in pressure as the bubble size increases. This is achieved in British "adjusted" uranium by adding small quantities of aluminum and iron and suitably heat treating to produce a fine precipitate of UAl_6 and U_6Fe, both within the grains and on the grain boundaries[5].

In uranium dioxide, the behavior of the inert-gas bubbles differs from that in uranium, and the large temperature gradients influence the migration and complicate the overall behavior.

A fuel element can conveniently be divided into three regions, with temperatures in the ranges 800 - 1500 C, 1500 - 1800 C and above 1800 C. In the lower-temperature range (800 - 1500 C) most of the fission gas is retained in the uranium dioxide and at the lower temperatures the gas is probably diffusing as single atoms. Evidence obtained from transmission electron microscopy after a moderate irradiation of 2×10^{19} fissions cm^{-3} and a subsequent anneal at 1400 C shows that the majority of the gas is in the form of small bubbles or clusters of the order of 40-100 Å diameter[1]. The study of the gas released from lightly irradiated samples (10^{15} - 10^{18} fission cm^{-3}) suggests a discontinuity in behavior and indicates an internal trapping of the inert gas[6] consistent with these observations. At these temperatures, the migration of bubbles is likely to be slow so that little coalescence would be expected, and the bubbles would be small and numerous with little swelling or gas release.

In the range 1500 - 1800 C, the bubbles will be more mobile and migrate up the temperature gradient, with fewer collisions than in the case of bubbles in uranium where the motion is not likely to be unidirectional. Many of these bubbles will be trapped in the grain boundaries and will coalesce there[7]. If the bubbles become large enough for them to interconnect, then much of the grain-boundary gas will be released; any grain-boundary migration will enhance the release.

About 1800 C it is known that most of the inert gas is released relatively rapidly, and it has already been inferred[8] that this is a consequence of the migration, up the temperature gradient, of the large lens-shaped holes. These holes are believed to migrate more rapidly than the small bubbles which are consequently swept up and do not accumulate or enlarge to cause swelling.

At high burn-up the high concentration of gas bubbles can modify the situation in those regions where the gas accumulates without release. Here a situation can arise where the bubbles become sufficiently large to interconnect and permit a "breakaway" gas release such as has been observed in uranium dioxide irradiated to less than 1.5×10^{21} fissions cm^{-3} at centre temperatures of less than 1500 C[9]. This quantitative description is applied to gas-release results from several laboratories.

On the basis of these ideas, and by analogy with the swelling behavior of uranium, it may be possible to reduce the release of gas by limiting the migration of the gas bubbles. This may be done by inhibiting the processes whereby the bubbles move, or by reducing the temperature gradients which are believed to be the cause of their migration.

*Sponsor: F. Rough

1. WHAPHAM, A. D., to be published.
2. FROST, B. R. T., et al., "Radiation Damage in Reactor Materials," (I.A.E.A., Vienna, 1963), 219.
3. BARNES, R. S., and D. J. MAZEY, Proc. Roy. Soc., 275 (1963), p. 47.
4. BARNES, R. S., AERE Report, to be published.
5. BELLAMY, R. G., Inst. of Metals Symposium on Uranium and Graphite, Paper 8 (1962).
6. MacEWAN, J. R., and W. H. STEVENS, *J. Nucl. Mater.*, 11 (1964), p. 77.
7. BRADBURY, B. T., *Trans. Brit. Ceram. Soc.*, 62 (1963), p. 159.
8. MacEWAN, J. R., and V. B. LAWSON, Jr., *J. Amer. Ceram. Soc.*, 45 (1), (1962), p. 42.
9. DANIEL, R. C., et al., Report WAPD-263 (1962).

A film depecting the phenomena described in the preceding paper will be shown.

1548 (BNWL-1522-3) AEC REACTOR DEVELOPMENT AND TECHNOLOGY PROGRAMS. Technical Activities Quarterly Report, April-June 1971. (Battelle Pacific Northwest Labs., Richland, Wash.). Oct 1971. Contract AT(45-1)-1830. 61p. Dep. NTIS.

CARBON DIOXIDES—yields from radiolysis of carbon monoxide, water vapor, and helium mixtures, (E)
CARBON MONOXIDES—CO-H_2O-He, radiolysis of, yields of carbon dioxide and hydrogen from, (E)
GRAPHITE—radiation effects on dimensional stability of, fast neutron, (E)
HELIUM—CO-H_2O-He, radiolysis of, yields of carbon dioxide and hydrogen from, (E)
HIGH TEMPERATURE LATTICE TEST REACTOR—physics measurements on mockup MSBR and HTGR-type lattices in
HYDROGEN—yields from radiolysis of carbon monoxide, water vapor, and helium mixtures, (E)
MOLTEN SALT BREEDER REACTOR—reactivity worth measurements in mockup lattice of, material
NEUTRONS, FAST—effects on dimensional stability of graphite, (E)
PLUTONIUM NITRATES—criticality of solutions of, effects of gadolinium additions on, (E/T)
REACTOR FUEL ELEMENTS—performance of defected Zircaloy-2 clad mixed oxide, postirradiation examination of, (E)
REACTORS, GAS COOLED—physics parameters of thorium oxide (ThO_2)-plutonium oxide (PuO_2) fueled high temperature, (E)
—cooling systems of graphite moderated, radiolytic reactions in, (E)
REACTORS, GRAPHITE MODERATED—cooling systems of helium cooled, radiolytic reactions in, (E)
WATER VAPOR—CO-H_2O-He, radiolysis of, yields of carbon dioxide and hydrogen from, (E) (H.D.R.)

NSA, vol. 26, 1972

31765 BEHAVIOR AND PHYSICAL-CHEMICAL STATE OF FISSION PRODUCTS IN FUEL ELEMENTS OF PWR TYPE REACTORS. Bazin, J.; Jouan, J.; Vignesoult, N. (CEA Centre d'Etudes Nucleaires de Saclay, 91-Gif-sur-Yvette (France). Service des Elements Combustibles et Structures). Bull. Inform. Sci. Tech. (Paris); No. 196, 55-71(Oct 1974). (In French).

The behavior of fission products in fuel rods of PWR type reactors was investigated using the analytical techniques available in the hot laboratory. The study was carried out on fuel elements irradiated in conditions similar to those in commercial power reactors or in very different conditions, in order to test the fuel up to utilization limits. The fission products present in the fuel elements were identified and located. The various compounds and alloys they form between each other or with the fuel were determined particularly with respect to the oxide-cladding reaction, for which an endeavor was made to define the conditions in which this reaction occurs, in terms of the irradiation conditions. (FR)

NSA, vol. 31, 1975

Oxide-Cladding Reactions and Their Effect on Water Reactor Fuel Column Behavior, *Jean Bazin, Joël Jouan, Nicole Vignesoult (CEA-France)*

Studies carried out on experimental pins and assemblies irradiated in light-water reactors have made it possible to clarify the role played by reactions between cladding and fuel on the behavior of the fissile column; these reactions linked to the presence of fission products create local blockages of the fuel on the cladding. These blockages associated with the phenomenon of densification under irradiation cause the UO_2 columns to fragment and shrink in length.[1] Figure 1 shows an example of fuel column fractionation brought about by these two actions for a pin irradiated at a burnup of 20,000 MWd/MTU, at a mean power of 600 W/cm and undergoing heat cyclings. The burnup and linear power for a specific geometry are the parameters that govern the appearance and amplitude of the reaction zones, respectively.

Processing of the experimental irradiations undertaken by the CEA made it possible to plot a curve (Fig. 2) that clarifies the conditions under which the oxide-cladding reaction appears at burnups between 10,000 and 25,000 MWd/MTU. The linear portion represented on Fig. 2 is a simplification of an exponential variation in a reduced burnup area. This was shown by plotting a log diagram (τ) MWd/MTU = f (1/p) W/cm, both these parameters being directly proportional to the respective time and temperature of the reaction zone.

The nature of the elements, as well as their respective concentrations, coming into play in these reaction zones was determined by analyses. In general, contact between the fuel and cladding is via

1. a zircone layer always <10 μm thick adjacent to the cladding
2. a second layer adhering to the fuel and to the ZrO_2 coat at the same time. This sometimes two-phase layer (Fig. 1) contains U-Zr-Cs and oxygen. Figure 3 shows an example of the quantitative distribution of these elements in the most frequent case of a single-phase layer when powers are <500 W/cm.

In conclusion, an attempt may be made to determine a formation mechanism of this reaction zone for powers of 600 W/cm or less. Up to around 10,000 MWd/MTU, the concentration of fission products is negligible, the zircone layer generally is only visible micrographically; for these burnups, adherences are very infrequent and always extremely localized.

ANS TRANS., vol. 20, 1975

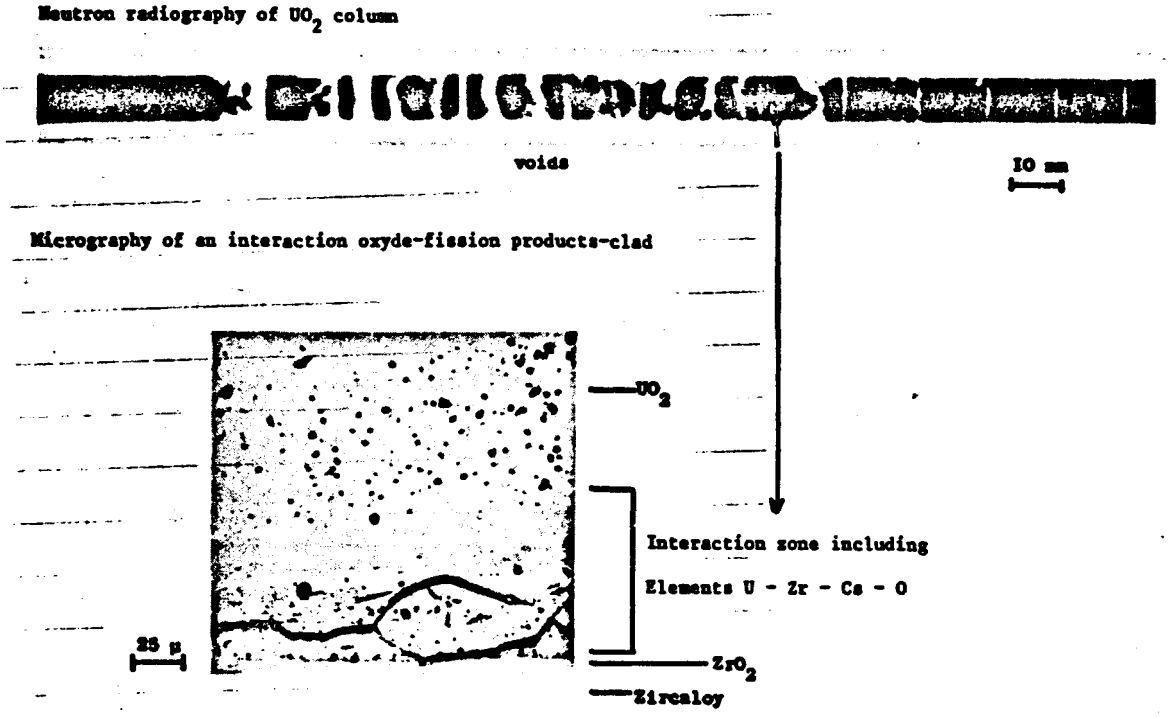


Fig. 1. Behavior of voids in fuel columns of a light-water reactor.

ANS TRANS., vol. 20, 1975

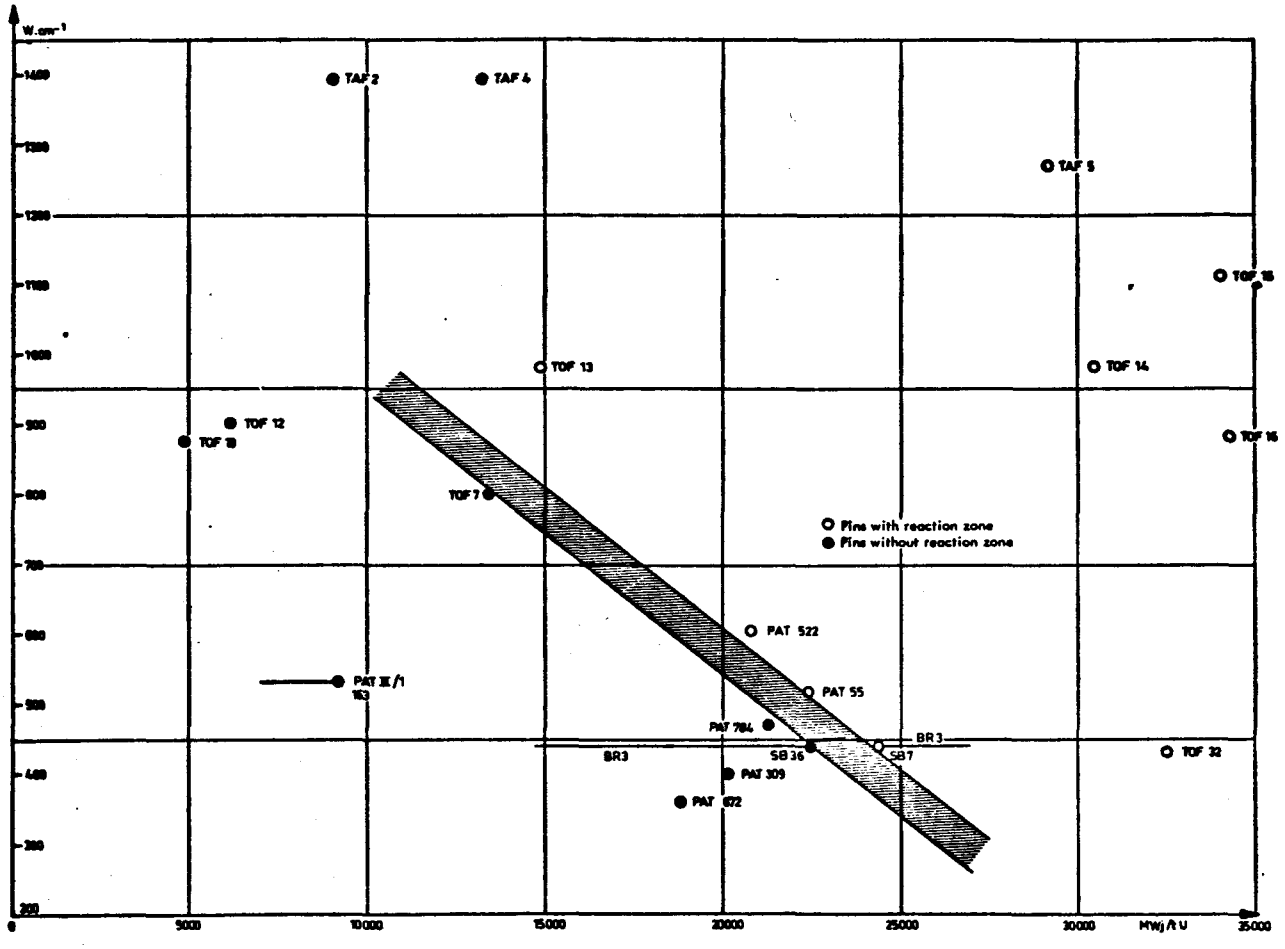


Fig. 2. Aspects of the oxide-cladding interface in terms of linear power and burnup.

ANS TRANS., vol. 20, 1975

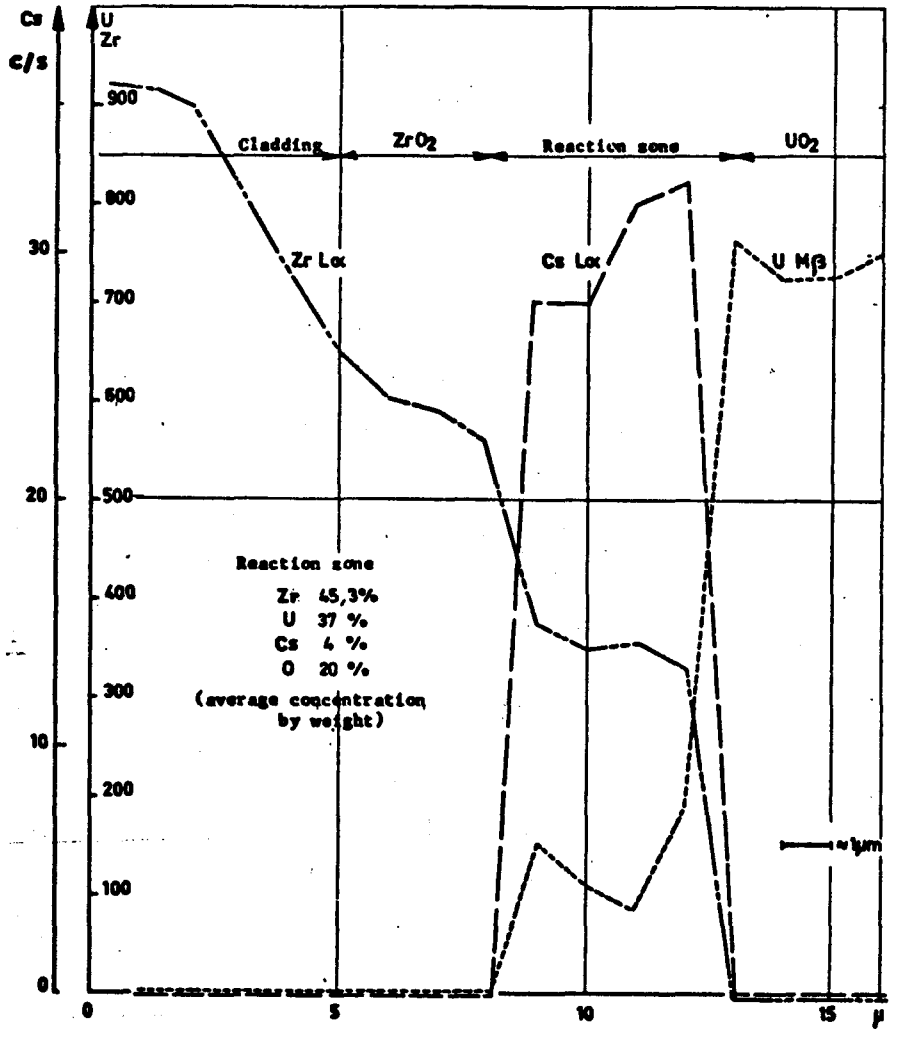


Fig. 3. Microanalysis of the oxide-cladding contact zone of TOF 32.

With higher burnups, fission products (particularly cesium) emitted by the fuel are in sufficient concentration to enter into the reaction; these fission products have a role whose importance increases with the power. The adherences provoked are then much greater and their effect on fractionation of the fuel column is all the more marked as the number of power cyclings grow.

1. G. DE CONTENSON, J. P. STORA, and N. VIGNESOULT, "UO_2 Densification, Irradiation Results, Laws and Models," ENS/ANS Conference, Paris (1975).

ANS TRANS., vol. 20, 1975

261010 Bobe, P.E. Nuclear Regulatory Commission, Washington, D.C. (USA). **Fuel performance of licensed nuclear power plants through 1974. NUREG-0032.** Jan 1976. v p. Available from NTIS: $5.00.

General aspects of fuel element design and specific design data for typical Pressurized and Boiling Water Reactors are presented. Based on a literature search, failure modes and specific failures incurred through December 31, 1974 are described, together with a discussion of those problems which have had a significant impact upon plant operation. The relationship between fuel element failures and the resultant coolant activity, radioactive gaseous effluents upon radiation exposure, plant availability and capacity factors, economic impact, and waste management, are discussed. An assessment was made regarding the generation, availability, and use of fuel performance data.
BWR TYPE REACTORS: fuel elements; FUEL ELEMENTS: performance; PWR TYPE REACTORS: fuel elements.

Atomindex, vol. 7, 1976

15645 ZIRCALOY-2 CORROSION IN STEAM-WATER MIXTURE UNDER REACTOR RADIATIONS. Cerrai, E.; Gadda, F.; Scaroni, A. (CISE, Milan). Energ. Nucl. (Milan); 18: No. 11, 607-14(Nov 1971).

Corrosion rates of Zircaloy-2 in steam-water mixture were studied in out-of-pile and in-pile conditions. Two closed-cycle loops were used for these tests. Most of the experiments were carried out in neutral condition, that is without any addition of chemicals to the feed water; only some preliminary tests were performed in an ammoniated environment. The experiments were carried out on corrosion coupons in as-pickled conditions; the exposure time for in-reactor tests was of the order of 70 days. The results obtained on both oxidation rate and hydrogen pickup are reported for the different experimental conditions investigated. (auth)

NSA, vol. 26, 1972

13643 BEHAVIOR OF ZIRCALOY-UO_2 FUEL ELEMENTS WORKING AT LOW TEMPERATURE (300°C). Chagrot, M.; Ringot, C.; Vidal, H. (CEA, Saclay, France). pp 91.1-91.4 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004.

Some irradiation tests have been performed to verify the behaviour of an "applied" sheath in UO_2 - zircaloy fuel element, when there is an interaction between the fuel and the sheath due, for example, to a power increase. The ELO tests had three purposes: determine the failure mechanism, in view of making possible counter measures, select the more favourable cladding material, measure the strains resulting from power increases during normal reactor operation. The main results have shown that several rupture mechanisms exist, one being purely mechanical, and that it seems that a cladding having a high initial ductility has a better behaviour. (auth)

NSA, vol. 32, 1975

15614 (CONF-7309129-1) INTERGRANULAR STRESS-ASSISTED CORROSION CRACKING OF AUSTENITIC ALLOYS IN WATER-COOLED NUCLEAR REACTORS. Cheng, C. F. (Argonne National Lab., Ill. (USA)). Aug 1974. 55p. Dep. NTIS $5.75.

From 5th European congress on corrosion; Paris, France (24 Sep 1973).

Corrosion-cracking incidents of structural components and fuel-element cladding in 30 water-cooled reactors have been reported. All pressure boundary penetrations have small seeping leaks, but pipe severance has not occurred. Failures of the non-pressurized boundary components have been limited to small fasteners and screws, none of which have constituted potential nuclear safety hazards, directly or indirectly, nor damage involving other components. The cracking incidents involving Types 304, 304L, and 347 stainless steel, Incoloy 800, and Inconel 600 were attributed to intergranular stress-assisted corrosion cracking associated with (a) oxygen, (b) caustic, or (c) hydrogen. Sensitization increases the susceptibility to cracking in dissolved oxygen, but it is not a factor in caustic solution. Other material conditions, corrective actions, and mechanisms of intergranular stress-assisted corrosion cracking are discussed. (49 references) (auth)

NSA, vol. 31, 1975

1. Ex-Reactor Biaxial Creep of Zircaloy-4 Cladding, *G. S. Clevinger (B&W)*

Creep tests of internally pressurized Zircaloy-4 tubing were conducted to generate data which are representative of the characteristic response of fuel cladding to temperature and stress. The analytical use of data reported in the open literature is complicated by the individualistic behavior of material produced by various manufacturing schedules, and by the broad variation in experimental conditions and testing procedures being used. By conducting tests on production quality tubes under optimized control of experimental variables, the characteristic deformation rate of the cladding for a given level of temperature and stress was determined. With systematic variation of the test conditions over a broad range of temperatures and stresses, a data base was developed which can be used to accurately predict changes in the deformation behavior of the cladding with temperature and stress.

Experimental conditions for the creep program included five temperatures and four stress levels. Temperatures ranged from 316 to 427°C (600 to 800°F) with maximum hoop stresses from 69 to 173 MN/m^2 (10 to 25 ksi). Tests are continuing and exposure times for some conditions have extended beyond 3000 h.

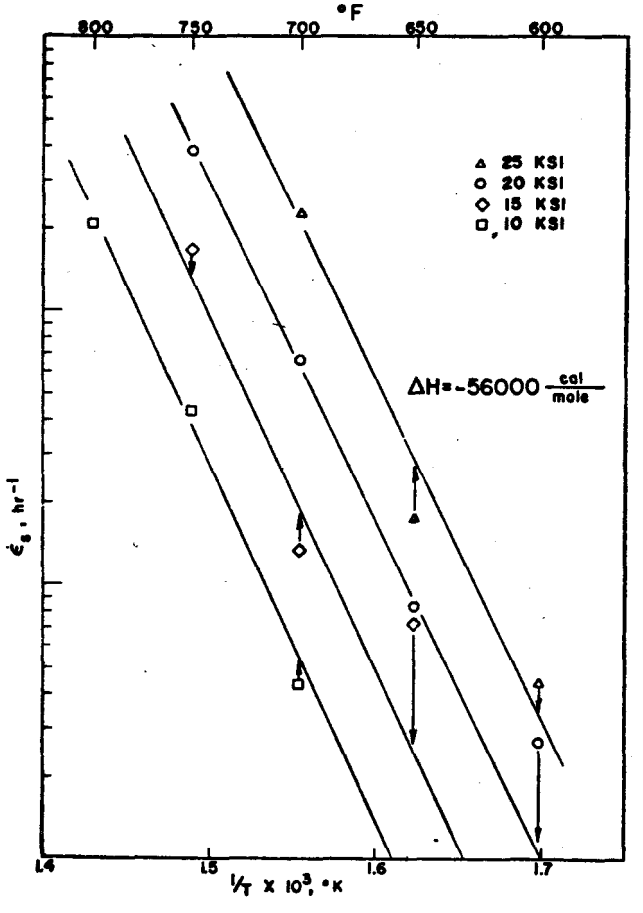


Fig. 1.

The experimental program was designed to generate creep strain versus time curves at a constant stress and temperature level for two lots of standard PWR cladding. The data generated in these tests indicate characteristic creep behavior for the higher levels of temperature and stress; that is, the strain versus time curves show the primary and secondary stages to be well defined and uniformly behaved. For the tests at a lower driving force (either low temperature or low stress or both), this behavior is not as well defined, and the transition from primary to steady-state behavior is more difficult to determine. Transient creep for these conditions extends to substantially longer times than might be expected, and may be complicated in some instances by time-dependent phenomena which exert an influence on the formation of the steady-state dislocation subcell network. The net result of this effect at long exposures and low driving force is to produce lower minimum creep rates than would be measured during shorter tests.

A simplified representation of the data collected thus far in the program is shown in Fig. 1. The secondary creep rates can be fitted reasonably well by a best set of parallel straight lines over all the conditions covered by the experimentation. Agreement with the Arrhenius-type relationship improves with long exposure times for low driving force conditions. This indicates that a consistent mechanism is controlling steady-state creep over this regime of temperatures and stresses.

The analytical use of the full creep curves incorporates sophisticated computer techniques described elsewhere.[1,2] The simplified engineering characterization of the stress and temperature dependence of the secondary creep rates indicates an activation energy of ~50,000 cal/mole and a power stress dependence close to 3. These values are broadly consistent with those reported in the literature, and are in agreement with classical considerations.

1. K. E. YOON and A. F. J. ECKERT, "Analysis of Fuel Cladding Collapse Due to Creep," *Trans. Am. Nucl. Soc.*, 19, 144 (1974).

2. D. E. CHESEBROUGH and H. E. WILSON, "Effects of Design Parameters on Collapse Time for Fuel Rods," *Trans. Am. Nucl. Soc.*, 19, 145 (1974).

CLADDING CORROSION IN MIXED OXIDE FUEL PINS

M.COQUERELLE and S.PICKERING

European Institute for Transuranium Elements, Postfach 2266, D-75 Karlsruhe

ABSTRACT

The corrosion of stainless steel and Inconel 625 cladding has been found to depend mainly on initial fuel stoichiometry, cladding temperature and burn-up. Corrosion increased as the initial O/M of the fuel was raised in the range O/M = 1.93 - 2.00; for an initial O/M = 1.93 corrosion was negligible. Depth of corrosion also increased as temperature increased in the range 500°C - 650°C; below 500°C corrosion was negligible even after 10 - 11 at. % burn-up. Depth of corrosion increased with increasing burn-up providing the cladding temperature was above 500°C. Increasing the Pu content of the fuel from 20 - 30 % had a negligible effect on corrosion. The various types of cladding material used, 1.4988 and 1.4970 stainless steels and Inconel 625 nickel based alloy, did not differ greatly in their corrosion behaviour.

The inner surface of stainless steel cladding has been found to become hardened and embrittled during irradiation and the amounts of corrosion products in the fuel/cladding gap after irradiation did not correspond to the amount of corrosion that had occurred. Because of non-uniformity of corrosion and mechanical fuel/cladding interactions it was not possible to describe corrosion in terms of simple chemical kinetics.

NUC. EN. MATUR., vol. 3, pt. 1, 1976

15781 METALLURGY OF ZIRCONIUM. Douglass, D. L. (Univ. of California, Los Angeles). At. Energy Rev.; Suppl. 1971, 1-470(1971).

A systematic exposition of the major problems relating to the metallurgy of Zr and its alloys, as dealt with during the period up to 1970 in over 700 references covering articles, reports, and conference and symposium proceedings is presented. The titles discussed include: crystallography; deformation and textures; physical properties; recovery, recrystallization, and grain growth; theory of alloying; phase equilibria; phase transformations and precipitation in alloys; mechanical properties; diffusion; radiation effects; corrosion; and oxidation. (719 references) (P.C.H.)

NSA, vol. 26, 1972

6. Shutdown Release of Fission Products in LWRs, *G. G. Eigenwillig, R. Hock (KWU-Germany)*

The amount of fission products, such as xenon, iodine, and cesium, released by defective fuel rods increases after shutdown.[1] We will attempt to explain this phenomenon, commonly called spiking.

In BWRs and PWRs, noble gases spike directly after a change of reactor power. Iodine and cesium spiking is remarkably different in BWRs and PWRs. In PWRs, spiking usually begins during power reduction; in BWRs, the major spike occurs some hours after shutdown.

To explain these differences, we investigated several shutdowns in more detail. We found that in a BWR, the major iodine/cesium spike occurs when the system pressure is reduced below 550 psi at shutdown condition.

There is no spiking if this pressure reduction occurs under power condition. This can be explained by the following mechanism: The coolant is at its boiling point at the surface. Below the surface, between the fuel rods, the pressure is increased by the pressure drop of the circulating water and the static pressure. This increase in pressure above the boiling point corresponds to a certain temperature difference to its boiling point. It is important that this temperature difference is higher at lower absolute temperature; e.g., at 100°C a pressure increase of 1 psi corresponds to 1°C subcooling; at 50°C it corresponds to 9°C subcooling.

Because of the (residual) heat flux, there is a temperature difference between fuel surface and coolant. This difference does not depend on the temperature level to any great extent since heat conductivity varies little with temperature. Thus, if water penetrates through a cladding defect at high temperatures, the increased fuel temperature is sufficient to bring the subcooled water to boiling, whereas it may not be sufficient, if the overall temperature level is lowered. The steam cushion will then collapse and liquid can leach the inner surfaces within the defective rod.

In a PWR, the degree of subcooling is much higher than in a BWR. Thus, in low heat flux regions of the core, water penetrating a cladding defect during power operation does not form a steam cushion. This possibility of leaching explains that at reactor power a PWR having the same noble-gas release as a BWR usually has a higher iodine release.[2]

In a PWR, power reduction leads to collapse of steam cushions that have been formed in parts of the core at higher power level. Thus, the iodine spike will appear during or immediately after shutdown.

In BWRs, in addition to the main iodine and cesium spike, a minor spike was sometimes observed shortly after shutdown. This spike, as well as the noble-gas spike, is explained by cracking of the pellets and release in a gaseous form.[3]

Our data reveal that about one-tenth to a few percent of the iodine and cesium inventory of defective fuel rods are released to the coolant during spiking. There are indications that about 90% of the iodine released from the pellets plate out on the inside surface of the fuel rods.[4] A comparison with the release during reactor operation showed that the spiking release is <30% of that plated out iodine.

In the case of iodine and cesium, the leaching process is more effective than any transport through a vapor phase at the same temperature. Thus, even months after shutdown the release rate coefficient of long-lived cesium isotopes (leaching) is only slightly below the values measured during reactor operation at much higher temperatures (gaseous release).

1. N. EICKELPASCH and R. HOCK, IAEM-SM-178/19 (1974).
2. R. HOCK and G. HECHT, AIM, International Meeting on Modern Electrical Power Stations, Paper 38, Liege, Belgium (1974).
3. R. M. CARROLL and O. SISMAN, "Evaluating Fuel Behavior during Irradiation by Fission Gas Release," ORNL-4601 (1970).
4. J. ALLEN, *J. Brit. Nucl. Eng. Soc.*, 6, 127 (1967).

ANS TRANS., vol. 23, 1976

25972 FISSION GAS YIELDS FROM POROUS SINTERED BODIES. Exner, H. E.; Moench, S.; Petzow, G.; Lukas, H. L. (Max-Planck-Institut fuer Metallforschung, Stuttgart). J. Nucl. Mater.; 43: No. 1, 8-12(Apr 1972). (In German).

Using a simple model the conditions for stationary diffusion of fission gases to open pores in porous reactor materials are quantitatively formulated. It is shown that in reactors with high fluxes the fission gases are able to escape from fuel elements with a surface temperature of 1200°C if the pore distances are similar to those obtained under normal sintering conditions. In porous uranium phosphide with 18% open porosity there is no danger of swelling caused by fission gases even at high fuel burnup. (auth)

NSA, vol. 26, 1972

60520 (AEEW-R-785(Pt.1)) LWR-WIMS: A MODULAR COMPUTER CODE FOR THE EVALUATION OF LIGHT WATER REACTOR LATTICES. PART 1. DESCRIPTION OF METHODS.

Fayers, F. J.; Davison, W.; George, C. H.; Halsall, M. J. (Atomic Energy Establishment, Winfrith (England)). 15 May 1972. 113p. Dep. NTIS (U. S. Sales Only).

LWR-WIMS is a comprehensive "modular" scheme of computation for studying the reactor physics aspects and burnup behavior of typical lattices occurring in modern Light Water Reactor designs. This report describes the basis of all the physics methods which have been incorporated in the code. The basic feature of the scheme is a coupled multi-cell spectrum calculation, which enables reduction of the energy group structure and the smearing of pin-cells containing fuel rods or poison pins. Both diffusion theory and transport theory modules in cartesian geometry are available for determining overall flux distributions in the reduced group structure. Special techniques are used to handle the positional effects of burnup and the buildup of fission product poisons. Some new approximate methods for dealing with highly absorbing regions in the context of diffusion theory, and for homogenizing the interior of the BWR heterogeneous control cruciforms are described. Various numerical studies designed to establish the accuracy of the principal aspects of the computational strategy are also discussed in the appropriate sections. 67 references. (auth)

NSA, vol. 26, 1972

14636 (INIS-mf-1121) BEHAVIOR OF URANIUM OXIDE FUEL CONTAINING BURNABLE POISONS. Flipot, A. J.; Delbrassine, A.; Gilissen, R. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). 1973. 13p. INIS.

From symposium on the results of five years of BR2 reactor utilization; Mol, Belgium (4 Dec 1973).

Incorporation of Dy_2O_3 and Gd_2O_3 as burnable poisons in the form of a uniform dispersion into UO_2 has been studied in view to increase the lifetime of fuel elements. Small amounts of impurities as aluminium or titanium oxide were found to accelerate the solid solution formation between UO_2 and Dy_2O_3 or Gd_2O_3. Irradiations tests have shown that the fuel with a burnable poison dispersion has a good stability and that the titanium oxide strongly improves the homogeneity of the fuel without adverse effect on the fuel behavior. (auth)

NSA, vol. 30, 1974

13061 INFLUENCE OF VARIOUS METALLURGICAL PARAMETERS AND NEUTRON FLUX ON CREEP OF ZIRCALOY 4 CLADDING TUBES. Frenkel, J. M.; Weisz, M. (CEA, Saclay, France). pp 90.1-90.6 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004-.

The creep strength of cladding tubes is one of the significant parameters in the design of fuel elements of light water reactors. Internal pressure creep tests were made on Zircaloy 4 tubes of some twenty batches differing in manufacturing method. It was noticed that creep strength varied quite widely with the annealing temperature and the cold work level of the tube. A high annealing temperature had a beneficial effect on the transverse creep strength, whereas the effect of high cold working was unfavorable. Completely annealed tubes produced the best results. This annealing effect is linked to an increase in the anisotropy of the material presumably resulting from modifications in texture. Internal pressure creep tests under neutron flux were made in a wide range of temperature and flux (330 to 385°C, 4×10^{13} to 2×10^{14} n/cm^2.s, E > 1 MeV). At high temperature and high flux significant creep acceleration was still observed. Even under flux, the effect of the annealing temperature on the transverse creep strength was very clear. (8 references) (auth)

NSA, vol. 32, 1975

40228 DEFECT PERFORMANCE OF UO_2–PuO_2 THERMAL REACTOR FUEL. Freshley, M. D. (Battelle-Northwest, Richland, Wash.). Nucl. Technol.; 15: No. 4, 209-28(Aug 1972).

The defect performance of Zircaloy-clad mixed-oxide fuel operating in a pressurized-water environment was studied from the standpoint of fission product activity release to the coolant and the effects of a defect on fuel behavior. The defect experiment described was conducted over a range of linear heat rating conditions including those which resulted in fuel melting to investigate the effects of different operating parameters. The results show that the activity release characteristics of the fuel rod are sensitive to the operating mode and that the major species released to the coolant are the fission product gases, iodine, the alkali metals, and tellurium. There were no localized dimensional perturbations associated with the cladding in the defect region although the ingress of steam vapor into the rod resulted in oxidation of the mixed-oxide fuel. Comparison of fuel structures formed in the defected and a companion nondefected rod indicates that within the experimental uncertainties associated with this experiment, the effect of defecting on the melting heat rating is less than 10 percent and possibly insignificant. An oxygen-rich atmosphere in the interior of the fuel rod during irradiation prevented massive hydriding of the cladding. (35 references) (auth)

NSA, vol. 26, 1972

40227 MIXED-OXIDE FUEL IRRADIATIONS IN THE PLUTONIUM RECYCLE TEST REACTOR. Freshley, M. D. (Battelle-Northwest, Richland, Wash.). Nucl. Technol.; 15: No. 4, 125-76 (Aug 1972).

The satisfactory irradiation performance of several types of mixed-oxide fuels suitable for the utilization of plutonium in thermal reactors was demonstrated by a significant number of experiments performed in PRTR. Heterogeneously and homogeneously enriched packed-particle and pellet mixed-oxide fuels were irradiated at peak linear heat ratings higher, i.e., >655 W/cm(>20 kW/ft), than those employed in the most advanced commercial reactors to significant burnups, i.e., >18,000 MWd/MTM. Early experience in PRTR provided some of the first data related to the effects of hydrogeneous impurities on Zircaloy clad mixed-oxide fuel performance. The demonstrated solution to the problem was achieved through the development of improved analytical and fuel fabrication techniques. Fuel rod cladding becomes oval shaped during irradiation due to creep collapse with the degree of ovality varying among the different fuel types. The fission gas release fraction for vipac mixed-oxide fuels varies linearly from a value of 0.05 for a volumetric average temperature of 600° to essentially 1.00 for a volumetric average temperature of 2250°C. Homogenization of the PuO_2 particles in mechanically-mixed UO_2–PuO_2 commences at fuel temperatures sufficient to cause sintering and equiaxed grain formation during irradiation. PRTR experience indicates that a fuel/clad reaction that is stoichiometry dependent occurs in Zircaloy-clad mixed-oxide fuels. The satisfactory defect performance of mixed-oxide fuels was demonstrated by several in-service and intentionally defected experiments although there are indications that defect performance may constitute an operating limit at very high linear heat ratings. (61 references) (auth)

NSA, vol. 26, 1972

26008 CORROSION AND HYDRIDING OF ZIRCALOY-2 AND OTHER ZIRCONIUM ALLOYS IN BWRs. Garzarolli, F.; Diez, W.; Francke, K. P.; Fricke, W. (AEG-Telefunken, Frankfurt am Main). pp 15-20 of Effects of Environment on Material Properties in Nuclear Systems. /Hurrell, M. L. (ed.). London: Institution of Civil Engineers (1971).

From International conference on the effects of environment on material properties in nuclear systems; London, England (1 Jul 1971).

Postirradiation measurements of oxide layers and hydrogen contents of Zircaloy-2 claddings of full size fuel elements of the German BWRs VAK, KRB, and KWL, which have been in operation for one to seven years, revealed that corrosion of the Zircaloy-2 was higher in-pile than out-of-pile. The hydrogen pick-up was only 3 to 7% of the theoretically possible value, and even after seven years of operation the net hydrogen content of claddings with 0.9 mm wall thickness was less than 100 ppM. In spite of varying heat fluxes of the fuel rods and different water chemistry of the reactors the over-all corrosion behavior of the claddings was very similar. The hydrogen pick-up seemed to be dependent on the oxide layer thickness and was locally higher at spots with spalled-oxides. Investigations of isothermal corrosion samples of Zircaloy-2, Zircaloy-4, Zr-Nb2.5 and Zr-Nb3-Sn1, which were inserted in the core of the VAK for one year, showed that the corrosion behavior of Zircaloy-4 is similar to Zircaloy-2, and that Zr-Nb2.5 has a very high in-pile corrosion rate in the oxygen-containing water of BWRs. The hydrogen pick-up of Zr-Nb3-Sn1 was lower than for the other alloys investigated. (auth)

NSA, vol. 26, 1972

237560 General Electric Co., Schenectady, N.Y. (USA). **Conditioning of nuclear reactor fuel.** (In Dutch). Het conditioneren van kernreactorbrandstof. **Netherlands patent document 7504391/A/.** Int. Cl. G21c3/00. 11 Apr 1975. 17 p. Priority 12 Apr 1974, USA; 10 figs.

A method of conditioning the fuel of a nuclear reactor core to minimize failure of the fuel cladding comprising increasing the fuel rod power to a desired maximum power level at a rate below a critical rate which would cause cladding damage is given. Such conditioning allows subsequent freedom of power changes below and up to said maximum power level with minimized danger of cladding damage. (Auth.).

FUEL RODS: fuel-cladding interactions.

Atomindex, vol. 7, 1976

31433 THEORETICAL ANALYSIS OF THE STRAINS PRODUCED IN NUCLEAR FUEL CLADDING TUBES BY THE EXPANSION OF CRACKED CYLINDRICAL FUEL PELLETS. Gittus, J. H. (United Kingdom Atomic Energy Authority, Springfields, Eng.). Nucl. Eng. Des.; 18: No. 1, 69-82(1972). (CONF-710903-14).

From first international conference on structural mechanics in reactor technology; Berlin, Germany (20 Sep 1971).

Some of the stress, force, and strain distributions produced in collapsed cladding by the expansion of cracked pellets during a power increase have been calculated analytically. As the radial and transverse cracks in the pellet open, the tendency for the cladding to stretch preferentially over them is reduced by frictional sliding at the pellet/clad interface. The frictional forces opposing sliding are intensified by a high coolant pressure (which holds the can onto the fuel) while the ability of the cladding to resist the friction forces without being locally deformed depends on its strength. The coefficient of friction, cladding dimensions, number of radial pellet cracks, strain hardenability, and temperature gradients around the cladding influence the tendency for cladding strain to be concentrated over the opening pellet cracks. (auth)

NSA, vol. 26, 1972

263401 Gittus, J.H. (UKAEA Reactor Group, Springfields). UKAEA Reactor Group, Risley. **Theory of the frictional interaction between nuclear fuel cladding and a cracked ceramic pellet. TRG-Report-2757(S).** Feb 1976. 42 p. Available from H.M. Stationery Office, price Pound1.50.

A summary is presented of the outcome of theoretical work detailed in five publications, reproduced as appendices, which is concerned with the tendency for the cladding tube of nuclear fuel elements to fracture as the result of power cycling or after a sudden upward power excursion. The relationship is shown between the properties of the clad, those of UO_2 pellets, and the tendency of the clad to fail during upward power excursions. The role of interfacial friction is explored and the benefit to be obtained by reducing it is calculated for cases where a soft metal interlayer is present. It is shown that the experimentally-confirmed magnitude of the strain-concentration in the arc of cladding over a radial pellet crack could not arise if there were interfaceons present. Accordingly, these defects, although they do occur in some sliding situations, are thought to be absent from the pellet clas interface in fuel pins. (author).

FUEL CANS: fractures; FUEL PELLETS: cracks; FUEL-CLADDING INTERACTIONS: sliding friction.

Atomindex, vol. 7, 1976

27992 (HW-67818(Rev.)) MULTIPLE RATE TRANSITIONS IN THE AQUEOUS CORROSION OF ZIRCALOY. B. Griggs, H. P. Maffei, and D. W. Shannon (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Dec. 1960. Contract AT(45-1)-1350. 24p.

The weight gain of Zircaloy-2, -3, and low nickel Zircaloy-2 during water and steam corrosion is shown to go through two or more repetitive cycles. It is assumed the weight gain curves truly represent the corrosion kinetics. Several corrosion mechanisms are considered in the light of the kinetics. No definite mechanism can be established as correct with the present data; however, a supposition that stresses in the film periodically cause it to crack at or near the oxide-metal interface appears consistent with the data. (auth)

NSA, vol. 15, 1961

16528 ZIRCALOY CLADDING MECHANICAL PROPERTIES. Hannerz, K.; Vesterlund, G. (Allmaenna Svenska Elektriska AB (ASEA), Vaesteraas). Nucl. Eng. Des.; 33: No. 2, 205-218 (Sep 1975).

Life-limiting aspects of the Zircaloy cladding material are discussed, with emphasis on the fuel-clad interaction type of failure. The tensile tests and creep properties and the changes of these due to neutron irradiation are reviewed. A section on high temperature properties of interest in loss-of-coolant accident (LOCA) analysis is also included. The paper is concluded with a discussion of the optimum choice of cladding material properties and the basis of the data reviewed. (NL)

NSA, vol. 33, 1976

15652 SOME CONSIDERATIONS CONCERNING THE IMPROVEMENT OF THE SCALING RESISTANCE OF ZIRCONIUM AND ZIRCONIUM ALLOYS. Hauffe, K. Werkst. Korros.; 22: 604-12(1971).

Improving the scaling resistance of Zr and its alloys requires a reduction of gas solubility in the metal and an improvement of scale adhesion to the base metal. For this reason only such alloy additions are eligible which are either incorporated in the scale layer or form a ductile or molten alloy phase between the base metal and the ZrO_2 layer being formed. In this way one would on the one side prevent the scale layer from becoming porous (thus also eliminating the possibility of hydrogen diffusion into the base metal) while on the other hand the scale layer would be prevented from spalling off under shearing stress. These possibilities are discussed in terms of probable mechanisms for the action of alloying additions such as Sn and Al. (Corrosion Abstr.)

NSA, vol. 26, 1972

15639 SOME CONSIDERATIONS ON THE IMPROVEMENT OF THE CORROSION PROPERTIES OF ZIRCONIUM AND ZIRCONIUM ALLOYS. Hauffe, K. (Univ., Goettingen, Ger.). Werkst. Korros.; 22: No. 7, 604-12(1971). (In German).

Problems arising in connection with an improvement of the corrosion resistance of zirconium alloys are discussed. In order to understand and clarify the corrosion mechanism in the scale layer, essential consequences of false arrangements in zirconium dioxide were studied. This is followed by an account of studies on the mechanism of the oxidation process, and based on these findings, a variety of methods is discussed which might lead to an improvement of the nonscaling property and corrosion resistance. Suggestions as to a new research program to be initiated for the study of nonscaling zirconium alloys are also mentioned. (INIS)

NSA, vol. 26, 1972

8565 IS ZIRCALOY AN ECONOMICAL CLADDING FOR FUEL ELEMENTS? Held, Ch. (Steirische Wasserkraft- und Elektrizitaets-AG, Graz). Atomwirt., Atomtech.; 16: No. 12, 632-3(Dec 1971). (In German).

Because of its superior neutron physical properties, Zircaloy replaced stainless steel as cladding material of fuel elements for pressurized water reactors. Experience showed, however, that Zircaloy cores develop defects ten times more frequently than stainless steel cores. A reevaluation of the two materials was performed from safety and economic standpoints. The study showed that, in spite of initial cost, long-term operation with stainless steel cladding may be more economical than Zircaloy and may be less prone to fission product release. (H.B.G.)

NSA, vol. 26, 1972

27399 (NP-19239) LIMITING ASPECTS OF FUEL ELEMENT PERFORMANCE IN WATER COOLED POWER REACTORS. Advanced Course Organized by the Netherlands-Norwegian Reactor School at Institutt for Atomenergi, Norway, 24-28 August 1970. (Institutt for Atomenergi, Kjeller (Norway)). 361p. Dep. NTIS (U. S. Sales Only).

The performance of UO_2 fuels and cladding materials in water cooled reactors is reviewed with emphasis on the various life-limiting mechanisms. Specific topics discussed include: UO_2 properties; corrosion and hydrogen pickup of zirconium base cladding; Zircaloy cladding properties; mechanical interaction between fuel and cladding; comparison of pellet and vipac fuels; fuel melting; heat transfer in reactor cores; fuel element design; and defective fuel element behavior. (H.D.R.)

NSA, vol. 26, 1972

2321 NMI-1191

Nuclear Metals, Inc., Cambridge, Mass.

THE AQUEOUS CORROSION OF ZIRCALOY CLAD THORIUM. S. Isserow. Sept. 20, 1957. 25p. Contract AT(30-1)-1565.

Attempts were made to improve the ability of Zircaloy-clad thorium to survive exposure of the core to water entering through small defects in the cladding. The failure of defected specimens was attributed to inadequate core-cladding bonds, which were not improved by an interdiffusion heat treatment. The thorium core may also be so weak as to be the point of failure in bond tests and to crack under the stress induced by the volume expansion on corrosion. Carbon additions hardened the thorium and lowered the corrosion rates. Aging treatments did not increase the hardness, but did increase the corrosion rates. The importance of the core-cladding bond is brought out by the successful protection of Zircaloy-clad uranium alloys against cladding defects. In this case, the interdiffusion heat treatment strengthens the bond sufficiently to permit compaction of oxide. (auth)

NSA, vol. 12, 1958

615 (UJV-2499-MCh) HIGH-TEMPERATURE TREATMENT OF SINTERED UO_2 PELLETS COMPACTED WITH BINDERS. Landspersky, H. (Ceskoslovenska Akademie Ved, Rez. Ustav Jaderneho Vyzkumu). [nd]. 28p. (In Czech). INIS.

UO_2 pellets compacted with 2% of polyvinyl alchohol or polyethylene glycol and sintered at 1500 to 1600°C were heat treated for ½ to 8 hours at temperatures between 1700 and 2200°C. At temperatures exceeding the sintering temperature neither polarization nor expansion was observed, the content of residual carbon after sintering being <50 ppM. Phase segregation was not found. Grain growth in these materials could be expressed by the equation $D^2 - D_0^2 = k.t^n$, where $n < 1$, but in most cases the values of n differed from the values observed in pellets compacted without binders. Similarly, the addition of binders resulted in a change in grain growth kinetics. Inhomogeneities in both grain size and pore size as well as in pore size distribution were observed that exceeded the inhomogeneities in pellets compacted without binders. A linear analysis applied in the quantitative estimate of porosity was found to be suitable for closed porosity, i.e., at high final pellet densities. The effect of micropores <1.3 μm could not be evaluated due to limitations resulting from the resolving power of the apparatus. (INIS)

NSA, vol. 26, 1972

27394 (BNWL-SA-3961(Pt.2)) LATTICES OF PLUTONIUM-ENRICHED RODS IN LIGHT WATER. PART II. THEORETICAL ANALYSIS OF PLUTONIUM-FUELED SYSTEMS. Liikala, R. C.; Uotinen, V. O.; Jenquin, U. P. (Battelle Pacific Northwest Labs., Richland, Wash.). [1971]. Contract AT(45-1)-1830. 87p. Dep. NTIS.

A theoretical analysis of UO_2-PuO_2 fueled, light-water-moderated lattice experiments has been performed to aid in establishing technical bases and design criteria for the utilization of plutonium bearing fuel in thermal power reactors. Results for UO_2 and Al–Pu lattices are included in order to understand the effects due to uranium and plutonium separately. The problems involved in calculating critical experiments are discussed. Estimates of the effects of various approximations inherent in the theoretical methods and/or analysis procedures are included along with the consequence on the results of the correlation. Uncertainties in the measurements and the neutron cross-section data are related to uncertainties in the calculated values of k_{eff}. Areas which should be investigated in future analyses are also identified. (84 references) (auth)

NSA, vol. 26, 1972

16525 REVIEW OF EXPERIENCE WITH WATER REACTOR FUELS 1968-1973. Locke, D. H. (UKAEA Reactor Group, Springfields). Nucl. Eng. Des.; 33: No. 2, 94-124(Sep 1975).

This review of water reactor fuel performance from 1968 to 1973 shows that defect levels in Zircaloy-clad UO_2 fuels have, at times, been up to 1 in 3 bundles/reactor yr or approximately 1 in 100 pins. Among the consequences of such events, are considerable efforts by manufacturers, designers and operators to understand the mechanisms of defects. As a result it has been possible to feed back the operating experience to give improvements in defect level to approximately 1 in 200 bundles/reactor yr or between 1 in 10^3 and 1 in 10^4 fuel pins. The various types of defects and limiting features of Zircaloy clad fuel have been classified and fitted to well established reliability technology. There is justification for establishing a statistical data bank on water reactor fuel performance, to establish models and confirm acceptable defect levels, similar to those available on reactor plant. Few defects can yet be described as "life limiting" in that incore lifetime is currently decided on economic grounds. Some potential life-limiting features have been identified during the review which should be the subject of further work to improve the long term performance of water reactor fuel. (NL)

NSA, vol. 33, 1976

4815 WAPD-PWR-CP-3166 [Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.]

THE EFFECT OF OXYGENATED WATER ON CLAD-AND-DEFECTED UO_2 FUEL SPECIMENS. J. M. Lojek and W. T. Lindsay, Jr. [June 1957]. 25p. $4.80(ph OTS); $2.70 (mf OTS).

Small pits were found in the surfaces of the pellets directly beneath the defects. The pattern of surface discoloration indicated that water had circulated through the annular gap between the pellet and the cladding of the test rods which contained 2 defects. The weight changes of the pellets were variable, but small. In 2 duplicate runs, the weight losses for the most severely attacked clad pellets were less than those for unclad pellets by factors of 84 and 12. It is concluded that defected cladding offers sufficient protective effect so that any air-addition accident of plausible extent and duration will not adversely affect deffected PWR fuel rods. (auth)

NSA, vol. 12, 1958

16895 (ORNL-4710) FINAL REPORT ON THE SECOND FUEL ROD FAILURE TRANSIENT TEST OF A ZIRCALOY-CLAD FUEL ROD CLUSTER IN TREAT. Lorenz, R. A.; Parker, G. W. (Oak Ridge National Lab., Tenn.). Jan 1972. Contract W-7405-eng-26. 57p. Dep. NTIS.

The second fuel rod failure experiment in the Transient Reactor Test Facility (TREAT) was performed with a seven-rod bundle of 27-in.-long, Zircaloy-clad UO_2 fuel rods in a flowing steam atmosphere. A water-reactor loss-of-coolant accident was simulated by operating the TREAT reactor at constant power for 30 sec so that fission heat in the UO_2 pellets caused the Zircaloy cladding temperature to rise 80°F/sec to a maximum of approximately 2400°F. The fuel rods were initially pressurized with helium to between 65 and 75 psia (77°F) to simulate accumulated fission gas. The Zircaloy cladding swelled and ruptured. The amount and distribution of swelling could result in the blockage of 91% of the bundle coolant channel area of a Boiling Water Reactor (BWR) at the location of maximum swelling. The average rod maximum circumferential swelling was 60%. Metallographic examination revealed ductile ruptures and significant oxygen pickup. Zirconium-steam reaction was 1.1%. The center rod was preirradiated to 2800 MWd/MT in the Materials Testing Reactor (MTR) and Engineering Test Reactor (ETR) to build up an inventory of fission products and to determine irradiation effects on fuel rod failure characteristics. No irradiation effect was seen on the swelling and rupture characteristics from this low-level irradiation. The release of gaseous fission-product ^{85}Kr from the irradiated center rod was approximately 0.5%. The release of volatile fission products ^{129}I, ^{131}I, and ^{137}Cs was slightly lower.

Approximately 2.5% of the ^{131}I released from the center rod was in a chemically unreactive form, probably CH_3I. (35 references) (auth)

NSA, vol. 26, 1972

4. Boron Stainless Steel Fuel Element Cladding in the Indian Point Reactor,* *C. R. Johnson (B & W)*

Fuel cladding in the first core of the Consolidated Edison Indian Point Reactor is AISI Type-304 stainless steel modified with 250 parts per million (parts/10^6) boron. Early development of this cladding material included melting small induction-furnace heats and several 10 000-lb electric furnace heats to gain experience for boron concentration control and to provide fabrication development material. No outstanding difficulties were encountered in manufacturing small-diameter, thin-walled tubing.

Mechanical property tests showed that the small boron addition did not significantly affect tensile strength, hardness and tensile ductility. Twist tests indicated appreciable decrease in ductility above 2200°F (of significance during hot working and welding operations).

Radiation effects on the room-temperature tensile properties were determined by testing material containing 300 and 500 parts/10^6 boron. Specimens were irradiated in the Materials Test Reactor (MTR) process water to a peak neutron flux of 9.24×10^{19} nvt (below 0.4 eV) and 4.01×10^{18} nvt (above 0.4 eV). Boron-10 depletion was about 8.6 and 14.3 $\times 10^{18}$ atoms/cm^3, respectively, in the 300 and 500 parts/10^6 material. The boron depletion in the 500-parts/10^6 specimens is equivalent to about 60% depletion of B^{10} in the 250-parts/10^6 boron material used for fuel cladding in the Indian Point Reactor. These irradiations satisfied the experiment objective of determining what effect small quantities of boron have on the extent of radiation effects on stainless steel. Irradiation has a slightly greater effect on boron stainless steel than on unmodified Type-304 stainless steel (Table I). However, the differences are minor and do not detract from the suitability of the boron alloy for fuel-cladding service.

Corrosion resistance of the boron stainless steels was determined in tests in 600°F borated water (1.2wt% H_3BO_3). Neutron exposures and boron depletion were about the same as those of the mechanical property specimens. The specimens were cold bent to form the 'U' and elastically stressed with a stainless-steel bolt through the legs of the 'U'. None of the specimens cracked, and there was no noticeable influence of boron content on corrosion. The results of these tests indicated that additions of up to 500-parts/10^6 boron to Type-304 stainless steel have no significant effect on corrosion of either irradiated or unirradiated material in 600°F borated water (Table II).

TABLE I

Room-Temperature Tensile Properties of Irradiated-Boron Stainless Steel

Boron content, parts/10^6	Thermal flux (<0.4 ev) 10^{18} nvt	Fast flux (>0.4 ev) 10^{18} nvt	Tensile strength, lb/in.^{2}	0.2 percent yield strength, lb/in.^{2}	Elongation, %
0	0	0	95 000	38 600	61.2
0	9.24	4.01	100 560	68 600	37.5
300	0	0	90 000	40 600	58.3
300	9.24	4.01	98 300	67 500	28.0
500	0	0	92 600	37 300	56.1
500	9.24	4.01	108 650	78 400	34.3

*Sponsor: W. K. Anderson

ANS TRANS., vol. 7, 1964

TABLE I

Saxton Thin-Walled 304 Stainless-Steel Test Results

Rod No.	Clad Wall[a] Thickness (Mils)	Diametrical Gap (Mils)	Unsupported Plenum Length (in.)	Calculated Maximum[c] Heat-Flux (Btu/h-ft²)	Calculated Power[c] kW/ft	Calculated Burnup[c] (MWd/MtU)	Calc Max Integrated Fast Exposure (nvt) $\times 10^{21}$	Deformation in Unsupported Plenum Area	Longitudinal Ridge
13	9.7	4	0.290	440 000	13	9500	1.2	Yes	No
14	9.7	4	0.265	440 000	13	9500	1.2	No	Yes
22	9.7	4	0.215	440 000	13	9500	1.2	Yes	Yes

[a]In all cases the cladding outside diameter was 0.380 in. with an O.D./T ratio of 39.2

[b]Fast flux >1 MeV

[c]At region of axial peak flux

It was concluded that the boron-modified stainless steel was suitable for fuel cladding in the Indian Point Reactor. The final cladding material, containing 250 parts/10^6 boron, has been in service for over 187 full-power days without incident. The operational experience with boron stainless-steel fuel cladding has justified the materials selection for the Indian Point Reactor and indicates that this material is suitable for fuel cladding in pressurized-water reactors in which a quantity of burnable poison is desired to help control excess reactivity at a diminishing rate over core life.

TABLE II

Postirradiation Corrosion of Boron Stainless Steel in 600°F Borated Water

Boron content, parts/10^6	Exposure time, h	Weight change - mg/dm²	
		Before descaling	After descaling
0	100	-34.8	-50.4
0	1 000	- 2.6	-13.9
0	1 000	+13.0	- 7.8
0	1 000	- 6.1	- 8.7
0	1 000	- 2.6	-13.9
300	100	- 2.6	- 7.0
300	1 000	+ 2.6	- 8.7
300	1 000	+ 1.7	-11.3
300	1 000	+ 5.2	-10.4
300	1 000	+ 7.0	- 8.7
500	100	- 0.9	-26.8
500	1 000	+ 6.1	- 8.7
500	1 000	- 1.7	-13.9
500	1 000	+ 9.6	- 9.6
500	1 000	+ 3.5	- 7.8

ANS TRANS., vol. 7, 1964

14. Cesium Interaction with Irradiated Oxide Fuel Pins,* *Irving Johnson, Carl E. Johnson (ANL)*

Fission product cesium may affect the in-reactor performance of fuel pins in at least four different ways. First, cesium may lead to the formation of low melting mixed-oxide compounds located at the oxide-cladding interface which facilitate the intergranular attack of the cladding. Second, cesium may react with axial blanket (or insulator) urania pellets to form cesium uranate which leads to swelling of the pellets and may cause rupture of the cladding. Third, cesium may react with fission product molybdenum to form cesium molybdate which, as a vapor species, can, under the radial thermal gradient in the fuel, transport molybdenum and oxygen through the unrestructured region of the oxide to the oxide-cladding interface and there participate in cladding attack. Fourth, cesium may form cesium uranate in the unrestructured region of the oxide and lead to swelling which would exert additional pressure on the cladding. In the present study, the factors that control the interaction of cesium with axial blanket urania pellets have been elucidated.

Out-of-reactor experiments in which cesium and hyperstoichiometric urania ($UO_{2.075}$) were heated together in sealed capsules at about 725°C for 75 h showed that cesium uranate is formed. The cesium partial pressure over mixtures of cesium uranate and urania was measured using a Knudsen effusion mass spectrometric method. The observed cesium pressures were found to be in good agreement with values computed using new thermochemical data[1] for cesium uranate and the assumption that the equilibrium reaction is

$$2Cs(g) + UO_{2+x} + \left(1 - \frac{x}{2}\right)O_2(g) = Cs_2UO_4(s) \ ,$$

where the oxygen pressure is determined by the stoichiometry of the urania. The calculated cesium pressure is equal to that in equilibrium with liquid cesium for nearly exactly stoichiometric urania; therefore, the reaction of cesium with hyperstoichiometric urania will reach equilibrium when the stoichiometry of the urania reaches 2.

For irradiated urania-plutonia, the reaction of cesium with fission product molybdenum to form cesium molybdate competes with the reaction to form cesium uranate. Using new thermochemical data[1,2] for cesium molybdate, it has been shown that the equilibrium cesium pressure over mixtures of Cs_2MoO_4 and urania-plutonia solid solutions of O/M ratio less than about 1.998 is greater than that over mixtures of Cs_2UO_4 and the mixed oxide. Thus, if the pre-irradiation O/M ratio of the fuel (mixed-oxide) pellets is much less than 2 (e.g., 1.97), there will be a tendency for cesium to migrate from the fuel pellets into the blanket pellets. If the latter pellets are initially hyperstoichiometric, then formation of Cs_2UO_4 may be expected which can lead to swelling and fracture of the cladding. However, if the initial O/M of the fuel pellets is close to 2, the cesium will tend to remain in the fuel pellets as cesium molybdate and interaction with the blanket pellets will not take place. These predictions are in general agreement with the observations of cesium attack on insulator pellets in irradiated fuel pins.

The volume change accompanying the formation of Cs_2UO_4 by the reaction of cesium with hyperstoichiometric urania has been used to derive the conditions of initial smear density and urania O/U ratio which lead to incipient pressure on the cladding. Urania pellets that have not been carefully protected from air oxidation typically range in O/U ratio from 2.10 to 2.15. For such pellets, pressure on the cladding would be expected if the initial smear density is greater than about 80%. If the initial O/U ratio of the urania is held below 2.05 and the smear density below 88%, the formation of Cs_2UO_4 should not lead to rupture of the cladding.

*Sponsor: Carl E. Crouthamel

1. P. A. G. O'HARE and H. R. HOEKSTRA, *J. Chem. Thermodynamics* (in press).
2. D. W. OSBORNE, H. E. FLOWTOW, and H. R. HOEKSTRA, Argonne National Laboratory, Private Communication.

ANS TRANS., vol. 17, 1973

2. Primary Hydride Failure of Zircaloy-Clad Fuel Rods,* *K. Joon (OECD-Halden)*

Primary hydride failures of Zircaloy-clad fuel rods are defects or failures caused by severe localized hydriding of the cladding as a result of contamination of the fuel with hydrogenous impurities. The most common contaminant is moisture. These failures appear both at low burnup (<500 MWd/ton) and after several thousand MWd/ton have been accumulated.

It is postulated that hydride precipitation requires a certain minimum ratio between the hydrogen flux at the surface and the diffusion velocity in the cladding. The latter depends mainly on the local temperature and temperature gradient, the former is proportional to the local partial pressure in the gas phase, which in turn is related to the average gas-phase hydrogen or moisture concentration. Based on this philosophy, data from test fuel irradiations in the Halden BWR were assessed. Some data from other sources have been included.

*Sponsor: J. M. Døderlein

The assessment indicates the existance of a failure/nonfailure boundary for primary hydride failures at an initial moisture equivalent concentration in the accessible hot void of 2.5 to 3 mg/cm^3 (see Fig. 1). This limit seems to be applicable over a wide range of irradiation conditions and rod design parameters. No such boundary was observed in relation to the total initial amount of moisture in the fuel rods. For an average fuel rod at the failure boundary, the total amount of moisture is 10 to 100 times higher than the amount necessary to produce a local hydride failure. The same is true for the resultant average hydrogen flux at the cladding i.d. surface compared to the minimum flux necessary to cause hydride precipitation.

A possible mechanism for primary hydride failure is that moisture (hydrogen), adsorbed by the fuel during production, is released during the first heating of the rod (hydrogen probably comes out at a later stage) and forms a supply reservoir in the colder parts of the void (plenum). At this stage part of the moisture is used for healing cracks in the ZrO_2 layer on the cladding i.d. surface. During irradiation the moisture is decomposed and oxygen and hydrogen are gradually adsorbed by fuel and cladding, respectively. Sites of localized attack are created most likely in a scratch due to fuel-cladding interaction, although a catalytic effect of fluorine cannot be excluded. Gas can be supplied from the reservoir through fuel-cladding gap, fuel cracks, and interparticle voids. If the gas composition is not suitable for hydriding when it leaves the reservior, it is made so under the influence of radiolysis during transport. A certain minimum local partial pressure is needed to cause severe hydriding. Due to the large excess in the reservoir, hydriding will then proceed to failure, unless the supply line is disturbed and apart from cases where the conditions are marginal. As a result of the gradual disappearance of hydrogen, sunburst formation will generally occur only early in the irradiation.

1. F. GARZAROLLI et al., *Tagungsbericht*, p. 549, Reaktortagung, Bonn (April 1971).

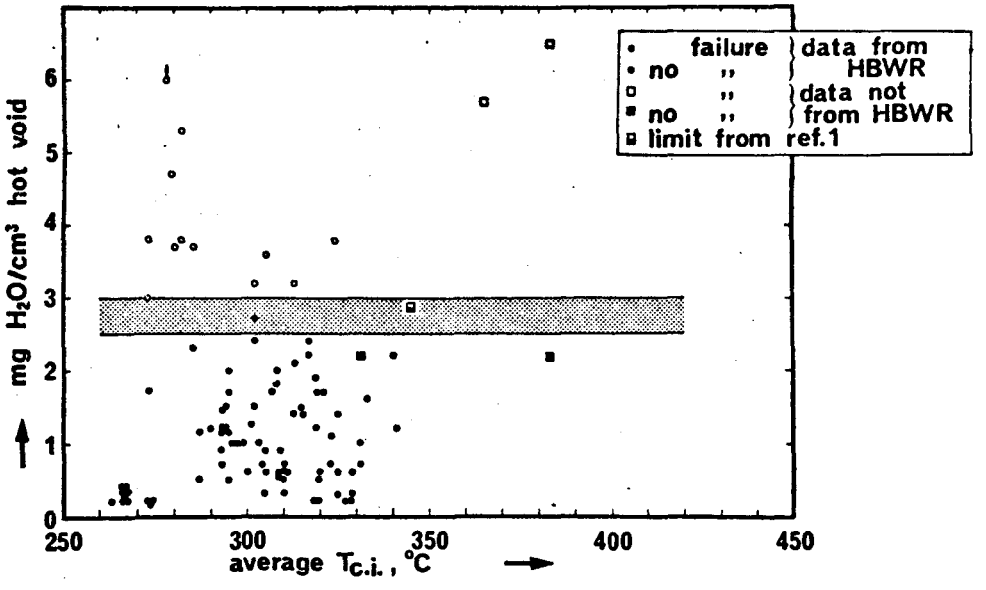


Fig. 1. Failure/nonfailure boundary for primary hydride failures in a plot of initial moisture concentration in the accessible void under irradiation conditions vs average cladding i.d. temperature.

5. High Burnup Fuel Performance in PWRs, *K. R. Jordan, S. Cerni, L. H. Boman (Westinghouse-NFD), invited*

As of the end of 1975, the cumulative electric energy production from Westinghouse-designed Zircaloy-clad PWR fuel exceeded 210,000,000 MWh. This experience was gained in 26 plants and represents an increase in generation of 84% during the previous 14-month period. This recent experience is given, updating Ref. 1. With few exceptions, fuel performance continued to be excellent.

Fuel burnup is given in Fig. 1. While much of this fuel performance experience has accrued as a result of new plants coming on line, a significant quantity of fuel has nevertheless achieved burnup levels consistent with its design bases. Of the 12 reactors refueled with Zircaloy-clad fuel since late 1974, four discharged fuel that had served through at least three cycles of operation.

An extensive in-reactor test program continues, to verify adequacy of current designs and to develop design improvements. Activities during the past year include seven TV examinations of fuel on site. Five of these included dimensional characterization of assemblies or removable test rods in addition to visual examinations.

The major increase in cumulative experience with Zircaloy-clad fuel has been achieved with no instances of defects due to moisture and cladding hydriding. Likewise, continuing surveillance by in-core instrumentation, detailed postirradiation examinations by fuel rod γ-scanning, and density determinations by metallography and pycnometry directly on fuel pellets have all confirmed that the problems associated with fuel densification and cladding flattening have been eliminated. In the last 12 reactor refuelings covering that period since the Fall of 1974, no fuel rods have been observed with flattened sections. There continues to be no observed instances of fuel defects due to corrosion.

Fuel rod bow continues to be observed in the low parasitic (LOPAR) fuel assembly designs. Observations made at recent refuelings have not demonstrated the expected reduction as a result of a design change, which provided space between the bottom of the fuel rod and the assembly bottom nozzle. An empirical projection of behavior has been developed from 27 sets of rod bow observations made on 22 regions of fuel of varying exposures up to that approaching the projected three-cycle duty. Reliability and safety evaluations based on the upper bound of this empirical projection continue to support the conclusion that there is no impact on core performance. These empirical modeling techniques, as well as the safety and reliability evaluations, have been described in a recent submittal to the NRC.[2] While the extreme bow case of rods in contact still has not been observed, thermal-hydraulic and wear tests, and analyses have provided confidence that rod contact is an acceptable condition and will not have impact on fuel rod reliability.

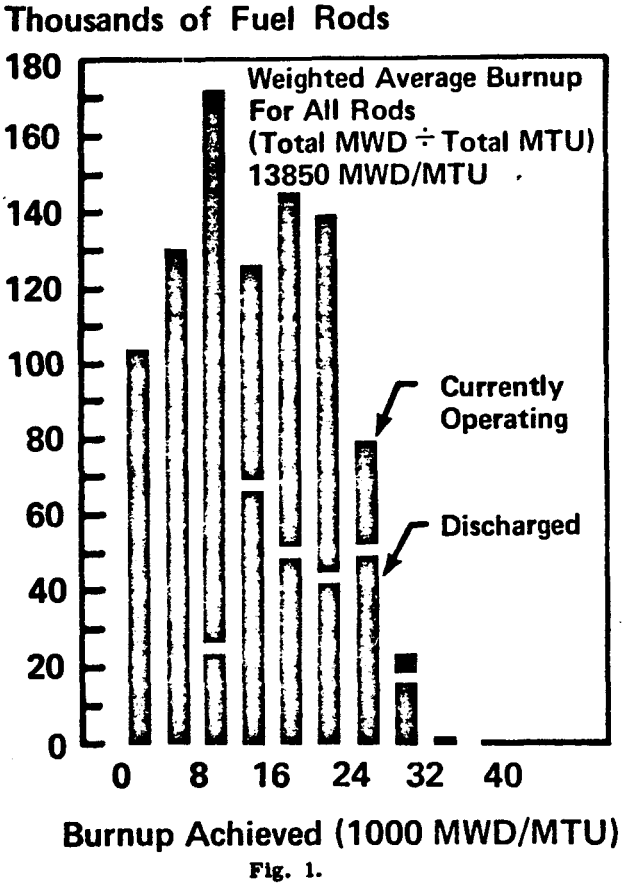


Fig. 1.

A low level of fuel startup defects occurred in one reactor prior to 1975. As a result of this event, startup limits of 3% reactor power increase per minute were recommended following refueling or extended reduced power operation. Since this recommendation in January 1975, 12 reactors have refueled and been brought back to power at the limited rate. One of these experienced a similar incidence of startup defects, possibly related to the amount of control rod usage. For the two reactors which experienced defects, subsequent operation was not restricted, coolant activity remained within acceptable bounds, and no assemblies were discharged as a consequence of the defects. Aside from these occurrences, fuel-cladding interaction defects have not been observed.

The reliability of Westinghouse Zircaloy-clad fuel continues to be high as significant high-burnup operation is accumulated.

1. F. W. KRAMER, "PWR Fuel Performance—The Westinghouse View," *Proc. Joint Topl. Mtg. Commercial Nuclear Fuel Technology Today*, ANS/Can. Nucl. Assn. (Apr. 1975).

2. J. R. REAVIS, W. J. LEECH, F. F. CADEK, S. CERNI, and J. M. HELLMAN, "Fuel Rod Bowing," WCAP-8892, Westinghouse Electric Corporation (1975).

A General Electric Fuel Performance Update,

Frank D. Judge, Harold E. Williamson, David T. Weiss (GE–USA)

I. INTRODUCTION

Over the past 15 years, General Electric has amassed a large fuel performance data base consisting of over 500,000 production fuel rods in over 10,000 fuel bundles, operating in nearly 30 boiling water reactors. We also recognize that reliable fuel performance is a necessity for achieving high plant availability and low energy costs. In the continuing quest to provide the best possible fuel, we have maintained an aggressive fuel surveillance program. Because of this program, when fuel problems are encountered we are able to effectively search out the affected fuel, determine the cause of the problem, and take corrective action.

In this paper we will attempt to summarize some of the facets of fuel and fuel channel performance we believe to be of current interest.

II. IT'S A NUMBERS GAME

With nuclear fuel, only a very few fuel failures can be tolerated and still achieve a high plant availability. A fuel rod failure rate in the range of 0.1 to 1.0% is undesirable. To demonstrate adequate performance throughout the fuel lifetime requires operating large numbers of fuel rods to both high power (kW/ft) and burnup (MWd/MT). For example, to expose these low-frequency failure mechanisms requires operating between 1000 and 10,000 rods in the power and exposure regime of interest. The defeat of a low probability mechanism can become a very frustrating statistical battle. In recent years, the rod moisture problem was an example where many thousands of data points were required to accurately establish the rod moisture distribution. It became necessary to eliminate a very small "statistical tail" to solve the problem.

We believe our vast data base of more than 500,000 fuel rods in service is an invaluable fuel performance resource.

III. SURVEILLANCE

Just having a large experience base is not enough; we believe it is very important to know what is happening in detail. To this end, GE has an aggressive fuel and channel surveillance and inspection program. Particular emphasis is put on the lead fuel and channels of each unique configuration (e.g., improved 7 × 7, 8 × 8).

It is not possible, of course, to examine all fuel at all plants with nearly 30 GE BWRs now in operation. But it is possible, by combining a lead bundle program with a program responsive to individual plant needs, to plan an effective surveillance program. A problem can be anticipated by monitoring plant operating data, especially the plant off-gas trends.

The GE fuel bundle is compatible with a fuel surveillance program in that an irradiated bundle may be readily disassembled underwater to permit individual rod removal and examination—and then be reassembled for further irradiation. By examining individual rods using nondestructive testing (NDT) and visual techniques, the ability to locate a defected rod is increased by a factor of about 50 over inspection techniques where the bundle periphery is inspected visually by television. To date, more than 37,000 individual production fuel rods have undergone nondestructive examination. Rods of specific interest can and have been transferred to the hot cells, such as the Vallecitos Nuclear Center in the USA, Grosswelzhein in Germany, or Kjeller in Norway for additional evaluations.

The channel surveillance program involves the combination of visual examination and dimensional measurements. To date, more than 100 different channels have been measured to 7 GE-BWRs; most of these have been given repeat measurements at later outages.

IV. RECENT FUEL EXPERIENCE

We have recently been through the difficult, and well-documented period of moisture- or hydride-related fuel rod failures. The cause of the problem was determined to be excessive moisture in fuel rods and action was taken to eliminate the problem. The solution being ultimately and predominately high-temperature vacuum outgassing of loaded fuel rods prior to final seal welding, and the addition of a hydrogen getter in the fuel rod to provide added assurance that no hydride-related failures will occur. That the problem has been eliminated is well demonstrated. First, by the excellent performance of 325 reload fuel bundles of the "improved"[a] 7 × 7 design which are loaded in 7 different GE-BWRs. These bundles have been tested by sipping after one cycle of operation. In these 325 bundles, only one leaker rod is suspected—that's one rod out of 15,925. And finally, three new large GE-BWRs, also predominately with improved 7 × 7 fuel, are well into their first operating cycle with very satisfactory fuel performance noted to date, as indicated by very low plant off gas.

Another fuel performance problem that has received a great deal of attention lately is pellet-clad-interaction (PCI). Where hydride failures can be generally characterized as occurring early in life, usually in the 0 to 11,000 MWd/MT range, and tending to occur somewhat randomly in the core and in a bundle; PCI failures tend to occur later in life, in the range 6000 to 27,500 MWd/MT, and can be a function of exposure, power, power change, design characteristics, and manufacturing variables.

Based heavily on work performed at Halden, where GE is an active participant, and at our own General Electric Test Reactor (GETR) and development laboratories, the mechanism was reproduced and changes to minimize the likelihood of PCI occurrence were determined. Changes were specified for both the cladding and the pellet geometry.

The cladding changes involved specifying a higher annealing temperature to achieve optimum uniformity of mechanical properties, and increasing cladding thickness to compensate for the lower yield strength of the new material. To minimize the potential for pellet ridging, a shorter, chamfered pellet with no dishing was specified. The first production fuel of this type was built during the second half of 1972 and is herein referred to as "improved" 7 × 7 fuel, to differentiate from earlier 7 × 7 fuel.

NUC. EN. MATUR., vol. 3, pt. 1, 1976

Because PCI is a statistical phenomenon occurring at high exposure, it will require the combination of large

[a]The term "improved" will be defined later in this section.

numbers of fuel rods operated to high exposure to demonstrate the effectiveness of the PCI fix. It is expected that the first "improved" 7 × 7 reload fuel bundles, put into service during the Spring of 1973 outages, will begin to provide meaningful data on PCI during their second cycle of operation. Hence, the tests and examinations performed during the Spring of 1975 outages will be of great interest. In the meantime, we are continuing to test fuel in Halden and GETR and are developing sophisticated analytical models to increase our understanding of the PCI mechanism.

With the introduction of BWR-6, GE is specifying an 8 × 8 fuel bundle. Since PCI failures decrease with decreasing fuel linear heat rate (kW/ft), it was determined to retrofit 8 × 8 fuel as reload fuel into operating plants and into not-yet-fabricated initial cores (BWR 4 and 5) to assure a high degree of fuel reliability and also to provide additional margin to thermal and thermal-hydraulic limits. The first 8 × 8 fuel was loaded, as reload fuel, during 1974. Consistent with the schedule for data feedback on "improved" 7 × 7 fuel, approximately two years will be required before statistically significant data will be obtained on this "improved" 8 × 8 fuel.

The benefit to fuel performance of operation at low linear heat rates has been well demonstrated on some of the early BWR-1s. A good example is the Garigliano initial core, which has operated to an average exposure of over 15,000 MWd/MT with only four leaker fuel bundles.

V. PLUTONIUM RECYCLE FUEL

General Electric has been involved with two significant development programs oriented toward providing a sound experience base for the orderly entry into the plutonium recycle fuel business. First, GE has been a participant in the Ente Nazionale per L'Energia Elettrica-sponsored plutonium development program in the Garigliano reactor. Second, we have participated in the Edison Electric Institute (EEI)-sponsored program, designing and building mixed-oxide fuel rods and bundles for irradiation in the Big Rock Point reactor.

We are now in the final stages of producing a fuel reload batch of mixed-oxide fuel bundle for loading in Garigliano during 1975. This batch of fuel will incorporate over 1400 mixed-oxide fuel rods. These rods were fabricated at Belgonucleaire, with uranium rod fabrication and bundle assembly taking place at Fabbricazioni Nucleari in Italy.

VI. FUEL CHANNELS

GE now has in service approximately 9000 Zircaloy fuel channels. As mentioned in Sec. III, the performance of these channels is being closely followed. Much useful data have been obtained. Using channel measurement data obtained from the production channels, we have established a channel creep model for use in predicting channel behavior. Of particular interest to BWR-6 owners is the performance of four pre-production 3.0-mm-thick channels now being irradiated in KKM. These channels, first loaded in January 1974, were given an interim inspection during the plants' August 1974 outage and were found to be performing in a completely satisfactory manner.

VII. CONCLUDING STATEMENT

General Electric has a large and growing boiling water reactor experience base with Zircaloy-clad pellet fuel. Also important, we have an aggressive surveillance program to permit us to obtain meaningful data out of this experience base. We have taken actions to eliminate, or at least severely restrict, the consequences of hydride and PCI failures. Of great significance is the fact that current performance requirements placed on the fuel are generally within the experience of statistically demonstrated fuel capability. This gives us a high degree of confidence that no new fuel problems will be encountered.

NUC. EN. MATUR., vol. 3, pt. 1, 1976

4816 WAPD-TM-97

Westinghouse Electric Corp. Bettis Plant, Pittsburgh. CORROSION TESTING OF ZIRCALOY-2 AND ZIRCALOY-3. S. Kass. Oct. 1957. 32p. Contract AT-11-1-GEN-14. $1.00(OTS).

Zircaloy-2 and Zircaloy-3 have found extensive applications in water cooled and moderated reactors due to their excellent corrosion properties in high-temperature water and steam. These excellent corrosion properties are inherent in the alloys; however, the corrosion behavior in high-temperature water and steam is greatly influenced by sample surface preparation and the manner in which the test is conducted. This report is concerned with the external factors influencing the corrosion behavior of the zirconium alloys. (A.C.)

NSA, vol. 12, 1958

10585 WAPD-T-416
Westinghouse Electric Corp. Bettis Plant, Pittsburgh.
CORROSION BEHAVIOR OF URANIUM–ZIRCONIUM ALLOYS IN HIGH TEMPERATURE WATER AND STEAM.
S. Kass and K. M. Goldman. Apr. 1958. 26p. Contract AT-11-1-GEN-14. $4.80(ph OTS); $2.70(mf OTS).

For presentation at 1958 Nuclear Congress, March 17-21, 1958, Chicago, Illinois.

The corrosion behavior of uranium–zirconium alloys containing 50 wt. % U has been determined in high temperature water at 600 and 680°F and in steam at 750°F. Data for other compositions in the range of 10 to 60 wt. % uranium are also presented to indicate the general effect of uranium content on corrosion behavior of uranium–zirconium alloys. It was found that, in general, the 50 wt. % U–50 wt. % Zr alloys are essentially unaffected by heat treatment. They exhibit a linear rate of corrosion at 680°F. These rates increase with increasing temperature according to the Arrhenius relationship. Corrosion life as a function of uranium content and temperature is presented and a mechanism of corrosion is suggested. The increase in total hydrogen content of the 50 wt. % U–50 wt. % Zr alloys resulting from corrosion at 680°F in water is presented. From a comparison of oxidation data obtained in water with those obtained in dry oxygen at similar temperatures, it is concluded that hydrogen is probably not involved in the mechanism of corrosion of these materials because of the good agreement between the two kinds of data. This idea is further substantiated by data from another laboratory which indicates that the hydride formed in the 50 wt. % U–50 wt. % Zr alloy is similar to $ZrH_{(1.2\ to\ 1.4)}$ rather than to uranium hydride. (auth)

NSA, vol. 12, 1958

38970 DIFFUSION COEFFICIENT OF HYDROGEN IN ALPHA ZIRCONIUM, ZIRCALOY-2, AND ZIRCALOY-4. Kearns, J. J. (Bettis Atomic Power Lab., Pittsburgh). J. Nucl. Mater.; 43: No. 3, 330-8(Jun 1972).

The diffusion coefficient of hydrogen in zirconium, Zircaloy-2, and Zircaloy-4 was determined in the temperature range, 275 to 700°C. No measurable differences were found among the coefficients for the three materials, nor was there observed any effect of grain size or cold work. However, small differences were found depending on sample orientation in rolled sheet. Values of the Arrhenius constants, D_0 and Q, for the longitudinal, transverse and through-thickness directions were, respectively, 7.73×10^{-3} cm^2/sec and 10,830 cal/mole, 5.84×10^{-3} cm^2/sec and 10,290 cal/mole, and 7.90×10^{-3} cm^2/sec and 10,730 cal/mole. These results, combined with an analysis of the crystallographic texture in the sheet, indicate that the ratio of the single crystal diffusion coefficients parallel and perpendicular to the "c" axis of the hexagonal structure, $D_{||}/D_{\perp}$, is greater than one, but probably not more than two. (auth)

NSA, vol. 26, 1972

9701 PROBLEMS ON NUCLEAR FUEL MATERIALS FOR WATER REACTORS. Kido, T. (Sumitomo Electric Industries Ltd., Osaka). Genshiryoku Kogyo; 17: No. 3, 5-10(Mar 1971). (In Japanese).

Development of fuel-cladding tubes made of Zircaloy is discussed. The quality of the Zircaloy is determined by the quantity of impurities, especially hydrogen and nitrogen. Zircaloy's crystals are hexagonal, and arrangements of crystals vary according to rolling and drawing methods. Strength of the Zircaloy is determined by direction of hydride orientation. Measurement of the hydride orientation can be made by circumferential elongation in internal pressure rupture testing. The fuel tubes with the desirable hydride orientation can be produced by wall thickness reduction method. The fuel tubes have an inside diameter tolerance of ±0.04 mm. Measurement of the inside diameter and thickness is made using air micrometers and supersonic thickness meters, respectively. As a defect detection method, supersonic waves are employed. The problems of fuel tubes for reactors are corrosion by high-temperature coolant water, hydride precipitation accompanying progress of corrosion, tube rupture because of pressure difference, swelling of fuel pellets in the cladding tubes, and fast neutron radiation effects. (Japan)

NSA, vol. 26, 1972

6. Trends in BWR Fuel Performance, *H. H. Klepfer, T. Trocki (GE-San Jose)*

Recent publicity on light-water reactor fuel problems (internal hydriding, pellet-cladding interaction, and fuel densification) can tend to overdraw their long-term implications. To properly assess the long-term implications, it is necessary to note important trends in both fuel and reactor system designs.

In BWRs, over 500,000 Zircaloy-clad UO_2 fuel rods have generated over 100 billion kilowatt hours of electrical power. Internal hydride attack of Zircaloy cladding has been the only notable cause of significant numbers of BWR fuel rod failures. Further, some cracking of cladding by pellet-cladding interaction has been observed. Definite corrective and preventive actions have been taken on both mechanisms.

To reduce hydride failures, a series of process changes was implemented, starting in early 1970. The earliest steps taken did not consistently reduce the frequency of occurrence of hydride failure to a satisfactory level. Therefore, additional corrective changes have been made. Increasingly definitive and more extensive quality measurements show these actions have decreased the probability of such failures by at least an order of magnitude. Manufacturing sequence and the chronology of reactor startup have not always been in the same order; but properly sequential analyses of fuel batch performance confirms the improvement trend. In addition, the BWR fuel produced since early 1972 has included perferential getters for hydrogen.

Cladding cracking has been observed in a small fraction of fuel rods after 2 or 3 cycles of operation. Investigation of this mechanism indicated that the cracking occurs in a fraction of fuel rods at high-peak thermal duty by excessive, highly localized, pellet-to-cladding interaction. Steps have been taken to reduce the probability of fuel failure by this mechanism. For fuel in production since early 1972, design changes have been incorporated; for fuel currently operating, special operating procedures have been recommended and implemented by the utilities. Importantly for assessing long-term trends, the entire BWR fuel and reactor system has been reoptimized to provide further margins in critical performance areas. For this new 8×8 fuel, the peak thermal duty has been reduced, providing margin to perforation under loss-of-coolant-accident conditions as well as improving fuel reliability. The 8×8 fuel design is now in production for reloads which will be loaded in early 1974.

Recently, in-core densification of UO_2 has received a great deal of technical attention. In PWR fuel, collapse of the cladding into gaps between pellets has been attributed to this phenomenon. Although densification of UO_2 has been measured in BWR fuel, and some gaps have been observed, no cladding collapse has occurred. The difference in fuel performance is due to several conservative features of the BWR fuel design including higher initial pellet density, lower cladding membrane stress, and lower creep rate.

Discussion of fuel cladding failure mechanisms inherently tends to emphasize failure aspects of fuel and this emphasis can distort perspective on overall reactor performance. Fission product leakage has affected power production to a varying extent in different reactors. In some cases, special outages have been taken for replacement of leaking fuel. More generally, fuel leakage has been accommodated with minor effect on planned power operation. The addition of off-gas treatment systems on some reactors has demonstrated dramatically reduced release of radioactive gases while permitting high power operation with leaking fuel. Off-gas systems are being added to essentially all existing reactors, and are included in all future BWRs. These systems provide assurance for controlling radioactivity released from leaking fuel, but the basic control will be in improved fuel reliability. The results of corrective actions taken to improve fuel reliability are becoming evident and their full impact will be realized in future fuel operation.

ANS TRANS., vol. 16, 1973

36566 (HW-75088) DIMENSIONAL VARIATIONS OF ZIRCALOY CLAD COEXTRUDED ON URANIUM FOR N-REACTOR FUEL. Interim Report. Knecht, R. L. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Aug 1962. Contract AT(45-1)-1350. 36p. Dep. NTIS.
Declassified 2 Nov 1971.
The common variations in clad dimensions on N-Reactor fuel are described. Studies to determine the causes and possible corrective actions to prevent these variations are also described. (J.R.D.)

NSA, vol. 26, 1972

8566 BEHAVIOR OF FUEL RODS AT HIGH BURNUP. Kummerer, K. (Kernforschungszentrum, Karlsruhe, Ger.); Stehle, H.; Weidinger, H. G. Atomwirt , Atomtech.; 16: No. 10, 545-52(Oct 1971). (In German).
Reported are problems of material as well as alterations and failures that occur at high burnups. Treated are fuel rods for boiling water reactors, pressurized water reactors, and fast reactors. Only oxides are discussed because of the fact that, presently, high burnup experience is available only for oxide fuel material. Physical and chemical changes in fuel and cladding, induced by radiation, are discussed. Specific problems of fuel rods are outlined from design to practical installation. (tr-auth)

NSA, vol. 26, 1972

4068 THE EFFECT OF A ZIRCONIUM STRENGTH DIFFERENTIAL ON CLADDING COLLAPSE PREDICTIONS. Lucas, G. E.; Bement, A. L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). J. Nucl. Mater.; 55: No. 3, 246-252(Mar 1975).

An investigation was made into the occurrence of a strength differential in the Zircaloy cladding of LWR fuels, and into the effect such a strength differential can have on the analytical predictions of cladding creep collapse during fuel densification.

The strength differential, or SD, refers to the difference in the compressive and tensile yield strengths of a material. It was concluded that an SD in Zircaloy cladding can have a significant effect on cladding collapse predictions; inclusion of SD considerations in cladding creep down analysis can increase predicted collapse times by a factor of two. (auth)

NSA, vol. 32, 1975

Pellet Cladding Mechanical Interaction, *J. E. Lunde (OECD Halden–Norway)*

Fuel cladding defects, like hydriding and collapse connected to densification, can be overcome by changes in specifications and fabrication processes without appreciably influencing fuel economy. No definite low cost solutions, however, have been established to avoid failures by PCMI.

It has been attempted to reduce the failure frequency by introducing operation restrictions to avoid large and rapid power increments, i.e., conditioning cycles, and by reducing the linear heat ratings through reduction in the fuel diameter. Further precautions have included interface lubrication and modified loading schemes.

Although such measures may partly eliminate the risks for PCMI failures, the mechanisms involved are not yet fully understood but should be further explored by carefully conducted experiments. Experimental data are needed, both for verification and modifications of performance models.

We will discuss the effects of the various measures mentioned on the basis of in-reactor measurements of axial and radial cladding deformation and fuel stack shortening (densification). The effects of design parameters, as well as operational variables like load-follow and power ramp (power shock) experiments, will be analyzed and illustrated by experimental data.

ANS TRANS., vol. 20, 1975

6429 ZIRCALOY CLADDING PERFORMS WELL IN PWR. B. Lustman, M. L. Bleiberg, E. S. Byron, J. N. Chirigos, J. G. Goodwin, and G. J. Salvaggio (Westinghouse Electric Corp., Pittsburgh). Nucleonics 19, No. 1, 58-63 (1961) Jan.

A destructive examination was made of Zircaloy-2-clad natural-UO_2 blanket fuel elements from the Shippingport PWR. The examination, plus operation and fabrication experience, increased confidence in zirconium-base alloys as cladding for UO_2-fuel rods. It was concluded that Zircaloy-2 cladding performed satisfactorily after a peak burn-up of ~9000 Mwd/ton UO_2; radiation exposure in 475°F water at 5.5 to 6.0 × 10^{21} nvt (fast) results in a marked elevation of yield and tensile-strength characteristic of fast neutron exposure of structural metals; local hydriding of cladding aggravated failure of defected elements only in in-pile hot-water loop tests; and the elimination of nickel delays or prevents hydriding failures. The only other type of failure encountered during the tests was "waterlogging," which is characteristic of unbonded fuel elements and can occur with any cladding material. Fatigue and tensile properties are tabulated for irradiated and unirradiated specimens of Zircaloy-2. Graphical representations are given of data on fast flux effects on 600°F tensile properties; notched impact tests; hydriding of defected fuel rods; and hydrogen absorption by Zircaloy-2 and -4 in 343°C water containing excess hydrogen. (B.O.G.)

NSA, vol. 15, 1961

23257 UO_2 PROPERTIES AFFECTING PERFORMANCE. Lyons, M. F.; Boyle, R. F.; Davies, J. H.; Hazel, V. E.; Rowland, T. C. (General Electric Co., San Jose, Calif.). pp 1-86 of Advanced Course on Limiting Aspects of Fuel Element Performance in Water Cooled Power Reactors. Kjeller, Norway; Institutt for Atomenergi (1971).

Uranium dioxide property data are presented and the effects of important variables such as density, temperature, stoichiometry, etc. are discussed. Property data are also discussed in relation to their role in various fuel life-limiting mechanisms which have been shown to be of practical significance for UO_2 containing fuel rods. These mechanisms are principally fuel-cladding interactions, fuel swelling, hydriding of Zircaloy cladding and fission product attack on stainless steel cladding. (Norway)

NSA, vol. 26, 1972

6. In-Pile Experience with Stainless-Steel-Clad UO_2 Fuel Elements in a Borated PWR Environment, *D. R. McClintock (WAPD)*

Types-304 and -348 stainless steels have demonstrated excellent behavior in a PWR environment. To determine the effect of exposure to a chemical shim environment, a study of the behavior of stainless-clad UO_2 fuel rods in the Saxton reactor was performed. This effort was part of a comprehensive irradiation program to proof test the effect of boron when added to the primary coolant in the form of boric acid. The program which is summarized in Table I, included full-size prototype fuel elements representing: 1) free standing cladding; 2) thin-walled cladding which operated with severe plastic deformation; 3) internally pressurized rods operating with biaxial tensile stresses in the cladding; 4) intentionally sensitized cladding; and 5) operation at high-power densities.

Fuel rods clad with cold-worked 304 stainless steel were operated in the Saxton reactor to peak power densities of 23 kW/ft. Neutron doses of 5×10^{21} n/cm^2 (>1 MeV) were experienced by the control-rod follower elements during operation of Cores I and II. Reactor operation was at 2000 lb/in.^{2} and 282°C coolant temperature. Stringent control of the primary-coolant chemistry was maintained with the boron concentration ranging from about 1600 to 300 ppm. Oxygen and chloride ion concentrations were less than 0.1 ppm. Hydrogen content was 300 cm^3/kg H_2O and the pH was controlled by the addition of an alkali such as lithium and potassium hydroxide.

Extensive postirradiation examinations were conducted in the Westinghouse PIF hot cells. Visual and dimensional examinations were performed, including profilometer measurement of the rods operated to 23 kW/ft. Samples of the fuel-rod surface deposits (crud) were obtained for chemical and radiochemical analysis. Several high-power density rods were punctured, the total gas volume determined, and the contained gases were collected and analyzed by mass spectrograph. Burnup and specific power output were derived from the average volumetric fission rate based on ^{137}Cs analyses of a rod section which included fuel and clad. Integrated fast-flux exposures were measured by wet chemical and radiochemical analysis of the stainless-steel cladding. Dosimetry calculations were performed based on the ^{54}Fe (n, k) ^{54}Mn reaction. Metallographic studies were made on samples of both the clad and UO_2 fuel.

Experience with cold-worked Type-304 stainless-steel-clad fuel elements operating to 23 kW/ft in the Saxton reactor, has shown no adverse effect of the borated environment on in-pile behavior. No evidence of failure, crack formation, or stress-influenced intergranular corrosion cracking was observed although the internally pressurized rods operated with up to 36 000 psi circumferential tensile stress on the outside diameter. No hideout of boron in the clad surface deposit (crud) or increased susceptibility to general corrosion or stress corrosion was noted.

TABLE I

Summary of Stainless-Steel Clad in Saxton

Test Assembly	Number of Rods	Material	Clad Wall (mils)	Peak Power Density (kW/ft)	Peak Exposure n/cm^2 (>1 MeV × 10^{21})	MWD/MTU
9-1	4	304 SS	23	23	1.4	11 000
10-1	12	304 SS	15	16	0.9	8 500
21	4	304 SS[a]	9.7	13.5	1.1	8 900
	1	304 SS[a]	9.7	13.5	1.3	10 000
22	5	304 SS[a]	9.7	5.5	0.4	3 200
24	1	304 SS[b]	15	13.5	0.3	3 000
25	1	304 SS[b]	15	13.5	0.3	3 000
27	2	304 SS[d]	15	5.5	0.5	4 000
	2	304 SS[d]	15	5.5	0.2	1 600
	3	16-20 SS	15	16	0.9	8 500
	3	348 SS[c]	15	16	0.9	8 500
Control-Rod Follower Elements	108	304 SS	22	8.2	5	42 000
Core	1 593	304 SS	15	13.5	3	26 000

[a]Collapsed Cladding
[b]Sensitized Cladding
[c]Annealed
[d]Internally Pressurized

27426 OUT-OF-REACTOR STUDY ON EXTERNAL-PRESSURE CREEP OF ZIRCALOY-2 FUEL CLADDING TUBES. Maki, H.; Hara, T. (Hitachi Ltd., Ibaraki (Japan). Hitachi Research Lab.). J. Nucl. Sci. Technol. (Tokyo); 12: No. 1, 43-52 (Jan 1975).

In Zircaloy-2 fuel cladding tubes the hour-glass shape of UO_2 pellets acquired by thermal distortion was simulated by flanges machined out at both ends of stainless steel pellets 21 mm long. The mock-up specimens thus formed were pressurized externally in a furnace, and the changes brought upon the tube diameter were measured at intervals. The external-pressure creep deformation was observed to proceed in a three-step diametral decrease, elliptical deformation, and final collapse. In the case of hollow tubes tested without pellets, elliptical deformation was observed, which accelerated with time until abrupt final collapse. Elliptical deformation was not observed on the pellet-filled tubes. Empirical equations were derived from the experimental results as functions of time, of hoop stress, and of temperature, to express the external-pressure creep strain behavior of the stress-relieved tube, pellet-filled and thus internally supported at intervals of 20 mm. No difference in the external-pressure creep deformation was observed between stress-relieved and recrystallized tubes under the condition of 350°C and 72.5 kg/cm^2. The pellet-filled specimens showed larger deformation than the hollow tubes in the process of diameter reduction. A certain length of unsupported distance in the range of 0~20 mm appeared to maximize the external-pressure creep deformation at the pellet center, under conditions similar to that of the present experiment. (auth)

NSA, vol. 32, 1975

41319 (GEAP-10371) ZIRCALOY-CLAD UO_2 FUEL ROD EVALUATION PROGRAM. Final Report, November 1967-June 1971. Megerth, F. H.; Ruiz, C. P.; Wolff, U. E. (General Electric Co., San Jose, Calif. Nuclear Fuel Dept.). Jun 1971. Contract AT(04-3)-189. 230p. Dep. NTIS.

The Zircaloy-clad UO_2 pellet fuel rods in Assembly SA-1 received their initial irradiation in the VBWR, and were subsequently exposed in Dresden-1 to an estimated average burnup of 8.2×10^{20} fissions/cm^3 (29,800 MWd/tU). The peak and maximum rod-average burnups were 10.4×10^{20} fissions/cm^3 (37,900 MWd/tU) and 9.3×10^{20} fissions/cm^3 (33,900 MWd/tU), respectively. The total operating time in both reactors was 1827 days. Local peak powers of 14 kW/ft were attained by some rods early in operation. No full-wall cladding penetrations were detected. The average external corrosion weight gain shown by the fuel rod cladding at the end of operation was 480 mg/dm^2, and the maximum hydrogen content measured was 146 ppM. Burst samples exposed to about 7×10^{21} nvt > 1 MeV showed hoop stresses at rupture of 5100 kg/cm^2 at 343°C and 4400 kg/cm^2 at 454°C. Strains to fracture of 2 to 8 percent were obtained with no clear temperature dependence indicated. Maximum fission gas release was less than 1 percent. The irradiated UO_2 was examined by light and electron microscopy and by x-ray diffraction. Full cross-section and radially variant fuel samples were measured for burnup, uranium and plutonium isotopic composition, neptunium, americium, curium, and certain fission products. Relatively high levels of ^{232}U, ^{236}Pu, and ^{238}Pu were measured on some samples. (auth)

NSA, vol. 26, 1972

2607 (LA-DC-12806) MEASUREMENT OF THE OXYGEN TO HEAVY METAL ATOM RATIO IN UNIRRADIATED MIXED-OXIDE FUELS. Metz, Charles F.; Dahlby, Joel W.; Waterbury, Glenn R. (Los Alamos Scientific Lab., N. Mex.). [1970]. 13p. (SM-149/33; CONF-711107-6). Dep. NTIS.

From Symposium on analytical methods in the nuclear fuel cycle; Vienna, Austria (29 Nov 1971).

The large amount of data that has been obtained by the different laboratories which have had extensive experience with the thermogravimetric method has led to the conclusion that differences as large as 0.02 in the O/M atom ratio are easily possible. These differences can be caused by variations in parameters such as the temperature of the adjustment treatment to a stoichiometric ($MO_{2.00}$) product, length of time of heating, and by the presence of moisture in the reducing gas. The pooled reproducibility of these measurements from seven laboratories is approximately 0.008, however, it has been observed to vary from 0.002 to 0.012 absolute standard deviation for individual laboratories. The accuracy, or bias, of this technique under a particular set of conditions is not known, a situation which arises from the lack of well-characterized materials with which to calibrate the methods. Recent information indicates that oxygen-deficient mixed oxides are unstable and tend to go slowly to $MO_{2.00}$ unless protected against oxygen. Failure to take this precaution in former comparisons might have been a significant factor affecting the values obtained. The greatest need at this time is to establish a sound experimental basis on which to answer the question of bias in any of the techniques used for this measurement. If this chemical property is as important as the analytical chemists have been led to believe, it is recommended that a study be made as to the feasibility of making available a set of characterized materials, standards if you will for purposes of method calibration and for an international sample exchange program. (auth)

NSA, vol. 26, 1972

4167 (LA-DC-12805) DETERMINATION OF NON-METALLIC SPECIFICATION IMPURITIES IN MIXED-OXIDE REACTOR FUELS. Metz, Charles F.; Waterbury, Glenn R. (Los Alamos Scientific Lab., N. Mex.). [1970]. 18p. (SM-149/34; CONF-711107-5). Dep. NTIS.

From Symposium on analytical methods in the nuclear fuel cycle; Vienna, Austria (29 Nov 1971).

Methods are described for measuring micro quantities of carbon, halogens, nitrogen, sulfur, phosphorus, and water in sintered mixed oxide fuels. These methods are capable of measuring these impurities in the parts per million range. An important source of bias in most of these methods is contamination from the laboratory. In addition to clean, orderly laboratories having good ventilation, constant vigilance by the operator is necessary to obtain sustained unbiased values. (auth)

NSA, vol. 26, 1972

Experience of Power Reactor Fuels in Japan,

Yoshitsugu Mishima (U of Tokyo–Japan)

Since the first commercial nuclear power station of the Calder Hall type for electricity generation began to operate in July 1966 at Tokai, owned by the Japan Atomic Power Co., an additional six LWR stations had been put into operation by September 1974, and the electricity generation by these seven has reached 3067 MW in total. Sixteen more LWRs of 13,509 MW electricity in total are now under construction, to be operated by 1979, and 60,000 MW is the target of nuclear electricity generation for 1985. Table I shows the seven stations operating as of September 1974.

Regulatory work on the actual construction and operation of electricity generating stations is to be carried out in Japan by the Ministry of International Trade and Industry (MITI). A committee on technological matters, consisting of experts from government research institutes and universities has been conducting advisory work. The safety examination of the power reactors has been done by cooperative work between this Advisory Committee and the Advisory Committee for Reactor Safety of the Atomic Energy Commission. In this report, the author would like to describe the experience on power reactor fuel in Japan through his experience as the member in charge of fuels and materials of the two committees mentioned above.

Some troubles have been experienced and the utilization factor of the electricity generation reactors in Japan is around 60% to date. The troubles reported to MITI number 56, as of March 1974, among which 44 are concerned with the reactor system, but none of them has had any effect on personnel outside the power station. The major troubles concerning fuel are the occurrence of leaking fuel rods, collapse of cladding tube, and fuel rod bowing in PWR.

Fuel assemblies have been examined by the Regulatory Body during fabrication as well as at the time of regular periodic examination of power reactors, which in Japan is to be done once each year. All the fuel assemblies taken out of the reactor core should be examined by leak testing through sipping and visual inspection, including observation through ITV, borescope, etc.

To reduce the number of leaking fuel rods, moisture control has been strictly carried out for both BWR and PWR fuel fabrication, as performed in other countries. Change of the outer geometry of the pellet to reduce ridging, which, in turn, delays the beginning of pellet-cladding interaction, has been considered, but the conclusion is somewhat different among BWR, PWR, and ATR designers on the different outer geometry to adopt; i.e., concerning chanfer and dish.

Cladding collapse was found in Mihama No. 1 (PWR) at the time of the periodic examination in 1973. It was found to have occurred in 20 assemblies, all of them belonging to Region No. 1 and containing pellets of 94% TD initial density without internal helium pressurization. All of them were removed from the core and no collapse was found in the periodic examination the next year. No unpressurized rod has been left in the present core. Densification and creep collapse have been discussed and some models are proposed. Experiments on both have been performed and part of the results have been reported to the Regulatory Body.

Bowing of fuel rods was found in September 1973 when regular periodic examination was carried out on Mihama No. 2 (PWR), during the visual test carried out in the spent fuel pool. Upon detailed observation, 16 fuel assemblies were proved to contain a bowed rod or rods and the minimum rod distance at the bowed center can be classified into (a) <1 mm (8 assemblies), and (b) 1 to 1.5 mm (8 assemblies).

The cause of this bowing was thought to be mainly attributable to the restraint of longitudinal differential elongation between the fuel rod and the control rod thimble tube, caused by irradiation. The temperature difference between the thimble tube and cladding tube containing fuel has much less effect. The cladding tube is cold worked and stress-relief annealed, whereas the thimble tube is fully annealed. A difference in irradiation creep rate of Zircaloy between these two kinds of heat treatment is well known.

Some mechanisms to explain bowing have been proposed both by the utility and fuel manufacturers, and the effect of rod contact due to severe bowing during operation was analyzed. The data to which the predicted rate of bowing should be fitted, however, are scarce and the Regulatory Body thought it still insufficient to accept any of those proposed mechanisms before further data are obtained. It is therefore thought to be more conservative to discharge the bowed rods, to eliminate any increase in the probability of causing leaking rods during the next period of operation. Of course they did not consider that a large accident could be caused by the continued use of such bowed fuel rods.

To reduce the accumulated restraint force by the spacer springs against differential expansion, it is better to use so-called bottom-off-type assemblies, in which a suitable clearance, say several centimeters, is kept between the lower end of the fuel rod and the end plate to allow the rod to elongate downward. But the fundamental remedy for the prevention of bowing will be found in reducing the holding force of the spacer grid spring to the extent necessary, as well as to make improvements in the design of the PWR fuel assembly to reduce the differential elongation due to irradiation creep between the thimble tube and fuel rod. These factors have been recommended by us for consideration to improve PWR fuel assembly design.

The next chance to observe the existence of any additional bowed rod or rods in Mihama No. 2 is January 1975, and the results will be reported at the Conference. This will give us additional data to decide which mechanism is to be accepted for the cause of bowing. Observation of the bowed rods after continued use for an additional year at Point Beach and H. B. Robinson II in the United States will also be taken into account for the decision.

TABLE I

Commercial Nuclear Power Station in Japan as of September 1974

Company	Plant	Location	Capacity (MW)	Construction Start	Operate	Reactor Type
Japan Atomic Power Co	Tokai No.1	Tokai-mura Ibaraki Pref.	166	Oct. 1959	Jul. 1966	GCR
//	Tsuruga	Tsuruga-shi Fukui Pref.	357	Apr.1966	Mar.1970	BWR
Kansai	Mihama No.1	Mihama-cho Fukui Pref.	340	Dec.1966	Nov.1970	PWR
//	Mihama No.2	//	500	May 1968	Jul.1972	PWR
Tokyo	Fukushima No.1	Futaba-cho, Okuma-machi, Fukushima Pref.	460	Dec.1966	Mar.1971	BWR
//	Fukushima No.2	Futaba-cho, Okuma-machi, Fukushima Pref.	784	Mar.1968	Jul.1974	BWR
Chugoku	Shimane	Kajima-cho, Shimane Pref.	460	Feb. 1969	Mar.1974	BWR
total			3067 (7 units)			

NUC. EN. MATUR., vol. 3, pt. 1, 1976

5. Interaction of Fission-Gas Bubbles with Structural Defects and Release from Nuclear Fuels, *F. A. Nichols, H. R. Warner (W-BAPL)*

It has been shown[1] that for ceramic fuel elements, fission-gas bubble motion due to thermal-gradient driving forces is much greater than random or Brownian motion under normal operating conditions, and occurs primarily by a surface diffusion mechanism. The force[2] on the migrating species which controls the rate of bubble movement may be expressed in the form

$$f = -\frac{Q_s^*}{T}\left(\frac{dT}{dx}\right)_{pore},$$

where Q_s^* = heat of transport for surface diffusion, T = absolute temperature, and $(dT/dx)_{pore}$ = thermal gradient in the pore. The velocity of motion has been shown to be[3]

$$V_s = \frac{2D_s \nu \Omega Q_s^*}{r k T^2}\left(\frac{dT}{dx}\right)_{pore},$$

where D_s = surface diffusion coefficient, ν = number of diffusing species per unit area, Ω = molecular volume, r = pore radius, and k = Boltzmann's constant.

A theoretical model has been developed for predicting gas release in operating ceramic fuel rods, using the concepts of bubble migration in a thermal gradient. The bubbles are assumed to 1) nucleate homogeneously and diffuse to dislocations, or 2) nucleate on dislocations. The bubbles are then constrained to move on the dislocation, where they grow by coalescing with other dislocation bubbles, are bombarded by bubbles nucleated in the bulk, or acquire individual gas atoms. The bubble can break free of the dislocation only if the force exceeds the dislocation line tension $\sim \mu b^2 \sim 10^{-4}$ dyn. The critical size to break free of a dislocation is thus

$$r_d = \left(\frac{\Omega T\, 10^{-4}}{2\pi Q_s^*\, dT/dx}\right)^{1/3},$$

where dT/dx is the macroscopic gradient in the matrix and related to the gradient inside the pore by $(dT/dx)_{pore} = \frac{3}{2}(dT/dx)$, assuming the pore conductivity << matrix conductivity.

The bubbles then move up the thermal gradient until they collide with a grain boundary. They are trapped on the boundary until they reach a second critical size given by

$$r_{gb} = \left(\frac{\Omega \gamma_{gb}\, T}{2 Q_s^*\, dT/dx}\right)^{1/3},$$

where γ_{gb} is the grain boundary tension.

Upon release from the grain boundary, the bubbles accelerate due to an exponential increase in velocity with temperature through D_s. Thus, a critical or threshold temperature exists at which motion first becomes appreciable.

An estimate of the gas release temperature from grain boundaries may be obtained as a function of exposure time by calculating the temperature at which V_s, using typical gradients, becomes sufficient to migrate $\sim \frac{1}{3}$ the pellet radius in the time chosen. All portions of the pellet above this temperature are assumed to have 100% release, since the bubbles may reach the fuel centerline and be released to the plenum (either immediately or during power transients).

In-pile fission-gas-release measurements[4] indicate that a significant portion of the released gas occurs as a result of power transients, presumably due to cracking. The cracks either directly intersect the bubbles or possibly cause stress-induced migration to cracks of bubbles in their immediate vicinity. Chalk River tests[5] designed for "carefully controlled irradiations," indicate that ~25% of the bubbles previously released from dislocations and residing on grain boundaries may be released by this mechanism in the intermediate temperature range between the critical release temperatures from dislocations and grain boundaries, respectively.

A third contribution, which dominates at low temperatures, is that due to recoil and knockout processes. This contribution is estimated to be no larger than 1% release/100 EFPD.

The sum of the three contributions, the total predicted gas release from UO_2, is shown in Fig. 1 for 4, 15, 100, 500, and 750 EFPD (Effective Full-Power Days), along with the experimental data of Lewis.[6] Thus, reasonable

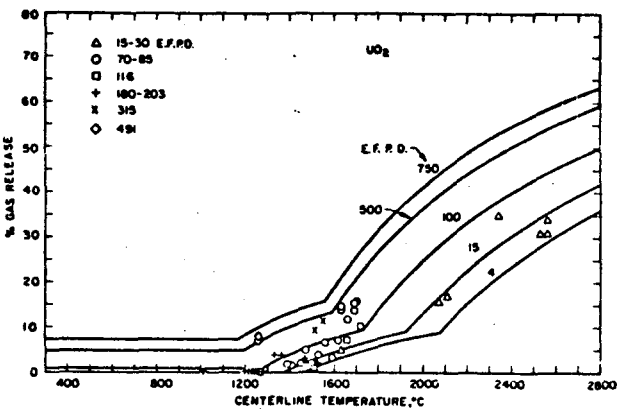


Fig. 1. Comparison of predicted and experimental fission-gas release from UO_2 for a thermal gradient of 10^3 °C/cm.

agreement is obtained, indicating the validity of the proposed models for predicting fission-gas release of operating fuel elements.

(Cont'd pg. 64)

1. R. S. BARNES and R. S. NELSON, "Theories of Swelling and Gas Retention in Reactor Materials," *AIME Conf. Radiation Effects*, September 8-10, 1965, Asheville, North Carolina, in press.

2. P. G. SHEWMON, "The Movement of Small Inclusions in Solids by a Temperature Gradient," *Trans. Met. Soc. AIME*, 230, 1134 (1964).

3. E. E. GRUBER, "Calculated Size Distributions for Gas Bubble Migration and Coalescence in Solids," *J. Appl. Phys.*, 38, 243 (1967).

4. M. J. F. NOTLEY, R. DESHAIES, and J. R. MacEWAN, "Measurements of the Fission Product Gas Pressures Developed in UO_2 Fuel Elements During Operation," AECL-2662, Atomic Energy of Canada Limited (November 1966).

5. W. BENNETT LEWIS, "Engineering for the Fission Gas in UO_2 Fuel," *Nucl. Appl.*, 2, 171 (1966).

ANS TRANS., vol. 11, 1968

42612 IMPROVEMENT OF THE RETENTION ABILITY OF COATED FUEL PARTICLES FOR SOLID FISSION PRODUCTS BY ADDITIONS OF SELECTED METAL OXIDES TO THE FUEL KERNEL. Nickel, H.; Foerthmann, R.; Hamesch, M. (Kernforschungsanlage, Juelich, Ger.). pp 634-7 of Reaktortagung, Bonn, 1971. Bonn; Deutsches Atomforum E. V. (1971). (In German).

From Reactor meeting; Bonn, Germany (30 Mar 1971). See CONF-710346.

The idea underlying the investigations was to improve the retention ability of oxide fuel material by adding small quantities of certain oxides which will form stable compounds with the solid fission products (e.g., spinels) and thus retain them in the fuel kernel. ZrO_2, Nb_2O_5 and Al_2O_3 additions were considered useful and have been tested in out-of-pile experiments. The results of the first series of experiments with Al_2O_3 additions include production of model fuel kernels, annealing of uncoated as well as of pyrocarbon-coated particles at various temperatures, and gamma spectrometric determination of fission gas release. (INIS)

NSA, vol. 26, 1972

6. Performance of UO_2 Fuel Elements to 10,000 MWd/MTU at High Powers,* *M. J. F. Notley (AECL), Invited*

Reactors fueled with natural UO_2 employ axial shuffling of short fuel bundles to improve the average fuel burnup. Additionally, economic considerations require high fuel power outputs, typically 700 W/cm length of single element. Thus, fuel thermal expansions and fission product gas releases are high, and since plenum voidage within the element is economically undesirable, internal gas pressures can become high enough to impair fuel sheath heat transfer or increase sheath strain.

The fuel performance under these conditions was assessed by running experiments to isolate individual effects, and by simultaneously developing a computational model to evaluate the many possible interactions (Fig. 1). Sheath strains, and, hence, fuel expansions, were determined both by postirradiation measurements and by use of strain gauges during irradiation. These experimental results were then compared with predictions based on a two-zone (plastic/non-plastic) model for fuel expansion to give information on the extent to which plastic flow would permit internal voidage to accommodate fuel expansion. This same model was used to calculate the voidage in cracks, etc., in the UO_2 pellet at power, as a stage in the calculation of internal gas pressures, which could be compared with measurements.[1]

*Sponsor: A. L. Lotts.

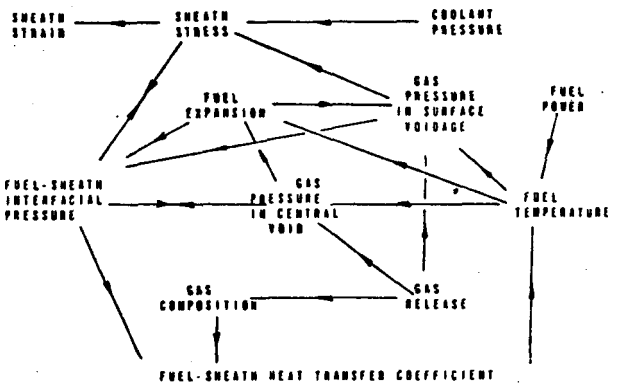


Fig. 1. Schematic diagram of the interdependence of variables considered in the program. (A ⟶ B denotes that a change in A affects B.)

Fission product gas retention in the fuel was correlated with temperature from analysis of microcores from irradiated fuel. These data were used to estimate steady-state gas release; the effect of power history was derived from measurement of the released gas from specific tests.[2] When the fuel power is changed, pressure measurements have shown a "step" release of gas which has been attributed to accumulation during steady-state operation of much of the released gas in the plastic core of the fuel. Calculations have shown that the volume occupied by this accumulated gas can be responsible for the anomalously large diametral expansions observed for an element irradiated with high internal gas pressures.[3]

Calculation of the fuel-sheath heat transfer is based on Ross and Stoute's analysis.[4] However, the results of element simulation calculations at high internal gas pressures have indicated a probable improvement of heat transfer due to creep of both fuel and sheath (alteration of solid-solid contact area) and the changes in nature of the contacting surfaces due to fission events in the fuel.

Predictions from computer simulation of element performance have been compared with experimental data, discrepancies between the two occasioning revisions of the program to eliminate the most probable source of the inconsistency. The analysis has pointed out the overwhelming importance, as far as predicting the performance at gas pressures in excess of coolant pressure is concerned, of the uncertainties in fuel-sheath heat transfer. Using reasonable assumptions, however, agreement between prediction and experiment has been generally satisfactory.

1. M. J. F. NOTLEY, "Calculation of Fission-Product Gas Pressures in Operating UO_2 Fuel Elements," *Nucl. Appl.*, 3, 334 (1967).
2. R. SOULHIER and M. J. F. NOTLEY, "Effect of Power Changes on Fission-Product Gas Release from UO_2 Fuel," *Nucl. Appl.*, 5, 296 (1968).
3. M. J. F. NOTLEY, R. DesHAIES, and J. R. MacEWAN, "Measurements of the Fission Product Gas Pressures Developed in UO_2 Fuel Elements During Operation," AECL-2662, Atomic Energy of Canada Ltd. (1966).
4. A. M. ROSS and R. L. STOUTE, "Heat Transfer Coefficient Between UO_2 and Zircaloy-2," AECL-1552, Atomic Energy of Canada Ltd. (1962).

2. Creep Collapse of Zircaloy-4 Cladding, *T. P. Papazoglou (B&W)*

The deformation behavior of externally pressurized Zircaloy-4 fuel cladding tubes was investigated at several temperatures and compressive stress levels related to PWR fuel operating conditions. The objective of the experimental program was to determine the creep collapse and creepdown rates of production-quality cladding and to provide the design data needed to verify analytical fuel performance codes. Since 1972, the problem of fuel densification has intensified interest in creep ovalization of cladding because of the potential for collapse into the voids formed in the fuel column.

Compressive creep deformation of externally pressurized closed tubes involves two types of stresses, acting in a nonuniform distribution around the tube circumference: (a) a hoop stress which causes creepdown, a contraction of the mean diameter of the tube, and (b) the bending or flexural stresses due to the nonuniform curvature or ovality on the tube surface. Thus, the study of creep collapse in real tubes presents formidable challenges because dimensional imperfections are neither fully characterizable nor uniformly distributed along the tubes. In the as-fabricated tubing, the stress distribution at a given transverse plane is further complicated by the presence of more than one major diameter (not a simple elliptical ovality) with variations along the tube.

This geometric complexity was shown to retard ovalization rate. An "incubation" period appears necessary for the weakest diameter to develop and predominate without further interference from other potential collapse diameters. Consequently, analytical calculations based on a simple elliptical geometry tended to conservatively underpredict the time to collapse.

The experimental specimens consisted of 5.75-in. lengths of PWR-size Zircaloy-4 tubing that were weld sealed while being internally pressurized with helium. Internal pre-pressurization served to achieve several levels of compressive stress (pressure differential) under a single constant external pressure of 17.24 MN/m^2 (2500 psig) at test temperature. Exact differential pressures were confirmed post-test by puncturing the test specimens and collecting the contained gas in a modified Sieverts apparatus. A variety of internal mandrels were used to simulate particular fuel operating conditions, including fuel column gaps.

The outer diameter of each individual specimen was measured by helical profilometry scans and plastic ovality was determined as a function of creep exposure time and specimen location every 100 to 150 h. Initial tube ovalities (o.d. max - o.d. min) were generally below 0.5 mil. Every scan included checks over two standard

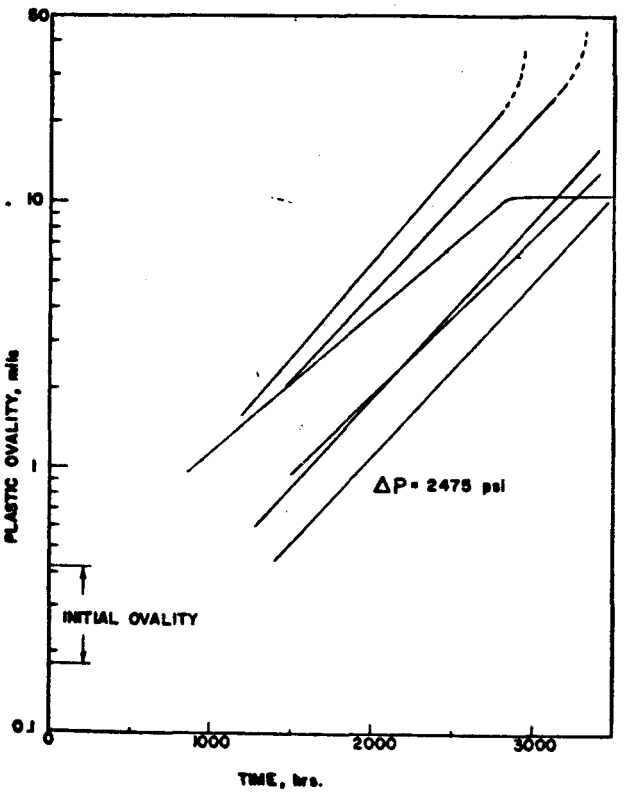


Fig. 1. Ovalization behavior of six typical specimens at 650°F.

diameters to ensure proper calibration of the instrumentation and an accuracy of ±0.1 mil. Initial wall thickness was determined ultrasonically and with a ball-contact micrometer. At each experimental temperature, a group of 45 specimens was simultaneously exposed to a constant argon pressure of 2500 psig. Specimens were periodically removed and scanned for dimensional changes. Figure 1 shows the ovalization behavior of six typical specimens with linear semilog plots beyond the initial incubation period.

The ex-reactor creep collapse kinetics of Zircaloy-4 cladding were thus determined as a function of temperature and compressive biaxial stress. Data from this study were supportive of a fuel design that ensures the maintenance of free-standing cladding throughout core life.

5. Fuel Performance in Boiling-Water Reactors/ *T. J. Pashos, H. E. Williamson, R. N. Duncan (GE-San Jose)*

Boiling-water reactor rod-type fuel-performance experience started with developmental Dresden-I prototype assemblies in the VBWR in 1957 and with full-scale central-station power reactor-fuel assemblies in Dresden-I in 1960. Since then, extensive performance experience has been obtained with rod-type fuel assemblies in an expanding number of boiling-water reactors, as listed in Table I.

The large quantities of production fuel required for first core loadings and subsequent reloadings have been made of UO_2 (or UO_2-ThO_2) fuel pellets clad with either Zircaloy-2 or Type-304 stainless-steel tubes designed not to collapse under the pressure of the primary coolant. The performance of Zircaloy-2-clad production fuel continues to be highly satisfactory as evidenced by lead assemblies in Dresden-I which started operation in 1960 and are continuing beyond the current (June 30, 1965) peak exposure of slightly over 20 000 MWd/t.

Failures in Zircaloy-2-clad production fuel have occurred only in the initial Dresden fuel. There have been 10 fuel-segment failures evidenced out of 77 000 segments of this type of fuel operated in Dresden-I. Continued operation to normally scheduled refueling outages after detection of these failures has shown that the failures progress slowly and plant operation has not been jeopardized. Half of these fuel-segment failures were caused by fuel-rod bowing resulting in localized corrosion from overheating, and the other half were clad cracks[1]. The absence of failures in Kahl-RWE fuel which has achieved a peak exposure about 16 500 MWd/t and in subsequent Zircaloy-2-clad production-fuel loads demonstrates the effectiveness of design changes made to eliminate these causes of failure.

Contrary to initial expectations, Type-304 stainless-steel clad fuels have proven unsatisfactory for boiling-water reactor service above peak exposures of 15 000 MWd/t. This is consistent with previously reported results[2,3] from developmental irradiation testing which started earlier and ran concurrently with the fabrication and initial operation of Type-304 stainless-steel-clad production fuel. Type-304 stainless-steel clad was found susceptible to intergranular cracking in at least some water-reactor environments. A number of stainless-steel-clad fuel rods have failed in service. However, these failures have not necessitated an early reactor shutdown for refueling.

The application of burnable poisons in production fuel loads for boiling-water reactors has been made in Elk River and Dresden-I. The Elk River first load has 600 parts/10^6 boron alloyed in the Type-304 stainless-steel clad. The second Dresden-I reload contained erbia, and the third Dresden-I reload contains gadolinia as burnable poisons. In addition, developmental fuel assemblies in Big Rock Point contain B_4C for axial power flattening and gadolinia as a burnable poison and for power flattening.

Concurrently a wide variety of developmental assemblies in small numbers have been inserted in various reactors to demonstrate performance of advanced designs for longer life, lower cost and or higher specific powers. These include alternate fuel forms, such as compacted powder, various clad materials (Zircaloy-2, Zircaloy-4, Type-304 stainless steel, Incoloy-800 and Inconel-600), various fuel-rod diameters and various clad thicknesses, as indicated in Table I. It was in tests such as these it was found that Type-304 stainless-steel clad cracks intergranularly around peak exposures of 6600 MWd/t for thin-wall collapsible clad and around 15 000 MWd/t for noncollapsible clad. The relative contributions of clad stresses and strains, material composition and impurities, coolant chemistry and irradiation effects to the clad cracking are under further investigation[4].

More advanced Zircaloy-2-clad fuel-rod designs now present in the Dresden-I core continue to operate satisfactorily at a current (June 30, 1965) peak exposure of 26 000 MWd/t. Metallurgical examination of irradiated Zircaloy-2-clad rods with exposures up to 875 days indicates the in-reactor corrosion rate approximately corresponds to corrosion in 700°F steam in ex-reactor tests[5]. However, the hydrogen pick-up fraction is extremely low, less than seven percent. Extrapolation of these data to five years of boiling-water reactor service indicates an average hydrogen content in the clad of approximately 150 parts/10^6, which is far below that which would adversely affect the properties of the clad during operation.

1. WHITE, G., "Developments in Boiling Water Reactors," A Conf. 28/P 205, *Third U. N. Intern. Conf. Peaceful Uses Atomic Energy* (1964).
2. NAYMARK, S., and C. N. SPALARIS, "Oxide Fuel Fabrication and Performance," A/Conf. 28/P/233, *Third U. N. Intern. Conf. Peaceful Uses Atomic Energy*, (1964).
3. PASHOS, T. J., et al., "Failure Experience with Stainless Steel Clad Fuel Rods in VBWR," paper presented at the ANS Winter Meeting, San Francisco, California, (November, 1964), *Trans. Am. Nucl. Soc.*, VII, 2 (1964).
4. PASHOS, T. J., "Investigation of the Cause of Failure of Stainless Steel Clad Fuel Rods in Water Reactors, First Quarterly Progress Report," GEAP-4915, (July, 1965).
5. BAROCH, C. J., et al., "Comparative Performance of Zircaloy and Stainless Steel Clad Fuel Rods Operated to 10,000 MWD/T in the VBWR," GEAP-4849, (July, 1965).

TABLE I

Operating Experience with Rod-Type Fuel in Boiling-Water Reactors

Reactor	Production Fuel				Developmental Fuel			
	No. of Rods	Clad	Fuel[b]	Approx Peak Exp 6 30 65 MWd t	No. of Rods	Clad	Fuel[b]	Approx Peak Exp 6 30 65 MWd t
Dresden-1 Type I	19 300[a]	Zr-2	UO_2	20 000	192	Zr-4	UO_2	22 000
Type III	10 300	Zr-2	UO_2	7 700	49[c]	Zr-2	UO_2	26 000
Type II	4 280	304 SS	UO_2	17 000	360	Zr-2	UO_2 Powder	700
	960[d]	304 SS	UO_2-ThO_2	17 000	108	304 SS	UO_2 Powder	16 000
					216	304 SS[e]	UO_2	13 000
JPDR	2 500	Zr-2	UO_2	2 000	72[c]	Zr-2	UO_2 Powder	2 000
Pathfinder[f]	7 800	Zr-2	UO_2	Initial Power Operation				
BONUS[f]	4 100	Zr-2	UO_2	Initial Power Operation				
Elk River	3 700	304 SS (600 parts/ 10^6 B)	ThO_2-UO_2	9 000				
Humboldt Bay	9 100	304 SS	UO_2	12 000				
Big Rock Point	10 100	304 SS	UO_2	7 500	363	Zr-2	UO_2	2 500
					363	Inconel 600	UO_2	2 600
					726	Incoloy[e] 800	UO_2 Powder	2 200
					363	Incoloy 800	UO_2	2 600
					484	304 SS[e]	UO_2 Powder	6 200
					484	304 SS[e]	UO_2	6 000
KAHL (Germany)	3 170[c]	Zr-2	UO_2	16 500				
SENN (Italy)	18 900	Zr-2	UO_2	7 800				
VBWR					800	304 SS[e]	UO_2 and UO_2 Powder	16 000
					400	Zr-2 Zr-4	UO_2	16 000

[a] Four segments per rod.
[b] Pelleted except as noted.
[c] Two segments per rod.
[d] Corner rods for power-peaking correction.
[e] Includes thin-wall collapsible clad rods.
[f] Boiler fuel of combined boiler-superheater reactor.

ANS TRANS, vol. 8, 1965

16527 INTERNAL CLADDING CORROSION EFFECTS.
Pickman, D. O. (UKAEA Reactor Group, Springfields). Nucl. Eng. Des.; 33: No. 2, 141-154(Sep 1975).

Internal hydriding has been the most persistent cause of defects in water reactor fuel pins. The mechanism of heterogeneous internal hydriding, the effect of bore surface condition, the critical amount of hydrogen, and criteria by which it may be specified are discussed. Production methods for achieving low hydrogen content are examined. Other factors which contribute to the initiation of a leak path through the cladding are discussed. Attack by other impurities, including fission products is considered, and the attack of stressed zirconium alloys by iodine is shown to be capable of producing major cracks. Recommendations are made on measures to minimize its incidence. (NL)

NSA, vol. 33, 1976

INTERPRETATIONS OF FISSION GAS BEHAVIOR IN REFRACTORY FUELS

R. L. RITZMAN, A. J. MARKWORTH, W. OLDFIELD, and W. CHUBB *Battelle Memorial Institute, Columbus Laboratories*
505 King Avenue, Columbus, Ohio 43201

Received November 12, 1969
Revised April 2, 1970

The swelling of the refractory nuclear fuels, UO_2, UN, and UC, at temperatures of the order of 1700°C, is related to the behavior of the fission gases as these gases make their way out of the fuels. In all three fuels, the fission gases first precipitate to form a two-phase system consisting of solid fuel and gaseous precipitates. These precipitates or bubbles grow in-reactor mainly by the accretion of new fission gas atoms. New gas atoms diffuse to the older bubbles (which are formed in the first few minutes of irradiation) causing these bubbles to grow and to swell the fuel. In UC, this process continues with very little change to produce rather large amounts of swelling. However, at about 10 vol% swelling of UC, bubbles begin to overlap to form channels and gas release begins to accelerate. In UN and UO_2 at about 1700°C, UN and UO_2 vapors begin to migrate across the growing bubbles as a result of the temperature gradients usually found in fuels in-reactor. This process, which is a form of zone refining, results in rapid and efficient gas collection and release. This gas release may reduce the stresses applied by UN and UO_2 to their claddings; but the zone refining process does not completely eliminate swelling stresses because swelling precedes gas release. Also, the zone refining or "restructuring" process tends to eliminate built-in porosity; so that while built-in porosity is very effective in limiting the swelling of UC, such porosity is relatively ineffective in reducing the swelling of UN and UO_2 at temperatures where restructuring is rapid. These processes are described in terms of classical chemical and physical metallurgical models; and the validity of the models is illustrated, although not necessarily proved, by the results of high-temperature irradiation experiments.

2. On the Pellet-Cladding Interaction Phenomenon, *J. T. A. Roberts (EPRI), E. Smith (Manchester Univ-UK), N. Furman (C-E), D. Cubicciotti (SRI)*

This paper reports on experimental and theoretical work sponsored by the Electric Power Research Institute (EPRI), which is aimed at the development of a detailed understanding of the pellet-cladding interaction (PCI) phenomenon. This is of current concern in UO_2-Zircaloy water reactor fuel since, in some instances where power changes have resulted in PCI, fuel-rod failures have resulted. In light-water reactors, PCI is now controlled primarily by accepting limitations on rate of reactor power increases, both overall and in local power shape. This remedy, while workable, is expensive in terms of lost plant output during slow increase to operating power; so there is strong economic incentive to gain a fundamental understanding of PCI which will lead to effective design or material-related remedies.

Experimental work on PCI[1] involves: (a) the detailed characterization of incipient cladding cracks induced by PCI in power reactor fuel rods; (b) the simulation in test reactors of the local power changes that cause PCI in fuel rods and the detailed measurement of subsequent cladding deformation; and, (c) the measurement in ex-reactor experiments of the appropriate critical properties that control both the driving force and response in PCI. The associated analytical work attempts to draw together the various data into a self-consistent model of PCI failure mechanism(s).

As part of an EPRI/C-E joint program, examination of fuel rods from Maine Yankee Core 1 has been completed.[2] This fuel was nonpressurized and contained UO_2 pellets which densified in-reactor. Fuel-cladding metallography and examination of the cladding internal surfaces revealed several features of relevance to PCI. A total of seven cladding cracks, varying in depth to 65% of the wall thickness, were located in three intact fuel rods coincident with the maximum power position. The cracks possessed the following characteristics: (a) they were associated with fuel pellet-pellet interfaces and usually located near to or opposite radial fuel pellet cracks; (b) they were tight (i.e., cracks that have small openings as compared with their length) and were normal to the cladding surface; and (c) they exhibited optical and scanning electron microscopic features consistent with a stress corrosion cracking (SCC) mechanism.

The incipient cladding cracks were only observed in fuel rods in which a relatively high fractional fission gas release (11 to 13.5%) occurred. These rods also exhibited fission-product redistribution to the cladding internal surface. (The peak burnup was ~15,000 MWd/MTU and the linear heat rating peak was 8 to 9 kW/ft.) Three major redistributed fission-product features were characterized by scanning electron microscopy: (a) linear deposits opposite to and outlining fuel cracks, and consisting of bands of uniform nodules (~2-μm diam) containing Cs and U, plus scattered crystals identified as a compound of Cs and I; (b) small localized glass-like deposits occurring opposite pellet-pellet interfaces and containing mainly U and Cs and occasionally Zr; and (c) randomly distributed circular mounds that had the same topographical features and elemental analysis as the deposits at the pellet interfaces. Occasional pieces of fuel were found cemented to the cladding with the same type of material as deposits (b) and (c). In two cases, fuel-cladding bonding was observed in the vicinity of the incipient cracks.

The results of this first fuel-rod examination campaign, when combined with the growing test reactor data base,[3,4] make a strong case for a mechanism of PCI failure which is chemically assisted. Consequently, current analytical work is proceeding to evaluate conditions under which cladding internal surface flaws might nucleate and/or propagate through the cladding normal to the internal surface by an SCC mechanism. Fuel-cladding bonding, the number of effective fuel cracks, interfacial friction coefficient, and K_{ISCC} for irradiated Zircaloy are the key parameters in the analysis. The analysis shows that inherent or nucleated cladding flaws on the order of 0.0003 in. can propagate by SCC if K_{ISCC} is very low or if the local stresses are raised by fuel-cladding bonding.[5]

1. "Planning Support Document for the EPRI LWR Fuel Rod Performance Program," 5R-25 (Dec. 1975).
2. "Joint CE/EPRI Fuel Performance Evaluation Program, Task C—Evaluation of Fuel Rod Performance in Maine Yankee, Core 1," Final Report (June 1976).
3. E. HOLSTAD and K. D. KNUDSEN, "Studies of Fuel-Clad Mechanical Interaction and the Resulting Interaction Failure Mechanism," A/CONF49/P/295; paper presented at *Fourth Int. Conf. Peaceful Uses At. Energy*, Geneva, Switzerland (1971).
4. S. AAS, K. D. OLHAUSEN, and K. VIDEM, "Fuel Failures Caused by Overpower Ramps, Nuclear Fuel Performance," BNS International Conference, London (Oct. 15-19, 1973).
5. E. SMITH, "Interaction Between UO_2 Pellet Cracks and Zircaloy Cladding in LWR Fuel Rods," Paper to be published in Proceedings of Symposium on Zirconium in the Nuclear Industry, Quebec (Aug. 1976).

1. The EPRI Light-Water Reactor Fuel Rod Performance Program, *J. T. A. Roberts, F. E. Gelhaus, E. L. Zebroski (EPRI), invited*

This paper will describe the objectives and scope and early results of the Electric Power Research Institute's[1] light-water reactor fuel rod performance program. This program is aimed at improving the reliability and mechanical performance of nuclear fuels by providing the industry with a better understanding of fuel rod failure mechanisms. The results of this work can be used to increase plant availability and capacity factors by reducing the extent of present plant operating restrictions.

The program consists of interrelated research at the fuel rod, fuel bundle, and whole core level, as shown in Fig. 1. Efforts in the fuel, cladding, and fuel rod properties area will provide the detailed measurement and correlation of performance and life-limiting mechanisms, and the improved understanding of those fuel and cladding properties necessary to predict and design for these characteristics. Analytical codes are being evaluated with future development to be based on improved phenomenological models; the analysis capability is benchmarked to significant data, using results both from separate effects tests (e.g., from the Halden and Studsvik reactors) and from controlled statistical tests in power reactors.

A significant quantity of power reactor bundle irradiation data will be accumulated and supported by pre- and postirradiation inspections and measurements. Both current and improved fuel rod design concepts will be investigated.

One important aspect of the work is to further understand both the mechanical and chemical nature of the pellet cladding interaction (PCI) failures that are currently impacting the operation of both BWRs and PWRs. To provide effective solutions to a problem such as PCI, it is necessary to quantify how the key variables (material or environment) synergistically control the fuel rod behavior. The available information, primarily derived from laboratory and test reactor studies, shows that cladding failure is likely to arise from a complex chemical-mechanical interaction whose origin could be in the fabrication, the radiation environment, and/or the duty cycle of the fuel rod.

Early results from the EPRI projects indicate a generic nature to those stress corrosion cracks observed in Zircaloy cladding. A high degree of commonality exists between cladding cracks induced in laboratory and test reactor experiments and those found in commercial power reactor fuel rods. The major features are (a) the presence of concentrations of fission-product species (Cs, I) at and near the flaw site, (b) the existence of incipient or part through-wall cracks originating from the i.d. surface, (c) the alignment of these cracks at 90 deg to the hoop stress, and (d) the macroscopically flat, brittle cleavage appearance of the crack surface.

Associated analytical work supports the view that stress corrosion will be a primary cause for failure under normal reactor conditions.

1. Planning Support Document for the EPRI Light Water Reactor Fuel Rod Performance Program, Special Report SR-25 (Dec. 1975).

36531 (HW-77602-RD) HYDRIDING AND THERMAL REDISTRIBUTION OF HYDROGEN IN N-REACTOR ZIRCALOY PROCESS TUBES. Shannon, D. W.; Westerman, R. E. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). 28 May 1963. Contract [AT(45-1)-1350]. 32p. Dep. NTIS.

Declassified 3 Nov 1971.

Zircaloy-2 corrosion and hydriding data from autoclave and in-reactor capsule experiments have been used to estimate hydriding rates of N-Reactor Zircaloy-2 process tubes. Thermal diffusion calculations show how local concentrations of hydride phase develop. Using the N-reactor temperature calculations for a maximum OD tube temperature of 360°C, the N Reactor process tubes were found to absorb corrosion product hydrogen at a rate up to 21 ppM/year, mostly due to OD corrosion. Lower OD temperatures were considered which result in lower hydriding rates. Hydrogen diffuses towards the inner cool wall and begins to precipitate a hydride "case" at operating conditions in about one to two years of operation at the higher temperatures. Subsequently, the hydride "case" on the inner wall is expected to grow, reaching a thickness of about 6 mils in 20 years, leaving the balance of the tube at 60 to 80 ppM hydrogen. This hydride case formation is due to the radial temperature gradients across the tube wall. Circumferential and longitudinal temperature gradients are shown to have little effect on hydride distribution except in the case of a severe local hot spot. The results emphasize the need for mechanical property studies of N-Reactor Zircaloy-2 process tubes containing a "case" of heavy hydride near the ID surface. (auth)

NSA, vol. 26, 1972

25954 (IJS-P-272) INVESTIGATION OF THE PROCESS OF BINDERS AND LUBRICANTS ADDITION AND REMOVAL IN CERAMIC NUCLEAR FUEL FABRICATION. Susnik, Dimitrij (Institut Jozef Stefan, Ljubljana (Yugoslavia)). Oct 1971. 38p. (In Slovenian). Dep. NTIS (U. S. Sales Only).

Thesis. Submitted to Ljubljana Univ., Yugoslavia.

The influence of organic additions, binders, and lubricants, on pressing, green strength, density, and microstructure of the UO_2 pellets are described. Best results are achieved with addition of stearic acid solution as binder and lubricant to the UO_2 powder. During presintering in CO_2 atmosphere, binders and lubricants decompose at about 200°C, whereas the rest of carbon reacts with hyperstoichiometric oxygen UO_{2+x} in CO_2 giving CO_2 and UO_2. Sintered UO_2 pellets were found to contain less than 30 ppM carbon. (auth)

NSA, vol. 26, 1972

2689 BMI-1387
Battelle Memorial Inst., Columbus, Ohio.
A VISUAL STUDY OF THE CORROSION OF DEFECTED ZIRCALOY-2-CLAD FUEL SPECIMENS BY HOT WATER. Elmer F. Stephan, Paul D. Miller, and Frederick W. Fink. Oct. 19, 1959. 32p. Contract W-7405-eng-92. OTS.

The failure of defected Zircaloy-2-clad uranium and uranium–2 wt. % zirconium fuel specimens in high-purity high-pressure water at 200 to 345°C was observed in a windowed autoclave. Time-lapse color motion pictures were taken to provide a record of the progressive changes ending in the complete disintegration of the core material in the specimens. Continuous measurement of the pressure increase caused by accumulation of hydrogen served to monitor the progress of the reaction when clouding of the water by corrosion products made visual observation impossible. The nature of the attack of all specimens was similar, although the time at which different stages occurred varied. Following an induction period, the first evidence of attack was the slow formation of a blister in the cladding area surrounding the defect. Eventually, a copious evolution of hydrogen occurred at the base of the swollen area. In general, a crack could be seen in the cladding at this stage. Catastrophic failure of the specimen followed swiftly. The time required for each phase of the reaction was reduced as the temperature was raised. Initial swelling occurred after about 24 min at 345°C but only after 8 hr at 200°C. Diffusion-treated uranium–2 wt. % zirconium-cored specimens were most resistant to attack. Specimens with beta-treated water-quenched natural-uranium cores were least resistant. (auth)

NSA, vol. 14, 1960

2636 (RCN-151) POST-IRRADIATION EXAMINATION AND MODE OF FAILURE OF THE DUTCH FUEL ASSEMBLY IFA-103, IRRADIATED IN THE HALDEN BOILING WATER REACTOR. van der Linde, A. (Reactor Centrum Nederland, Petten). Apr 1971. 119p. Dep. NTIS (U. S. Sales Only).

Three out of seven fuel pins in the IFA-103 test assembly failed after operation in the Halden Boiling Water Reactor. Post-irradiation examination of the longitudinally cracked, Zircaloy-2 clad fuel pins revealed that in two of the failed pins the tantalum tube, incorporated in the center of the vibrationally compacted fuel stack, was locally attacked. Moreover, one of these pins had a top weld which was deformed from the inside by a stainless steel spring during welding. The formation of Zr–Fe–Cr and Zr–Fe–Ni eutectics in the heat affected zone of the cladding resulted in hair cracks and a blistered ring on the outer surface. Although these phenomena contributed to the severity of the clad failures, the primary action resulting in the first clad fracture was attributed to a local fuel-clad reaction. It was found that at sites where a fuel particle was pressed firmly against the inner clad wall, oxygen and, when present in the pin interior, also hydrogen was quickly in the cladding. This phenomenon resulted in the formation of a sunburst; a spherical zone around the fuel-clad contact which was enriched in oxygen and depleted in hydrogen. This sunburst, as observed at room temperature, was surrounded by radially oriented hydrides. Absorption of oxygen resulted in local deterioration of clad ductility properties so that in combination with the increase in stress imposed on the cladding by the expanding zirconium lattice, small radial cracks were formed. Propagation of these primary cracks occurred via stress oriented, radial hydrides towards the outer surface up to a point where the remaining clad wall failed ductile by necking and 45° angle fracturing. The post-irradiation examination results are discussed in relation to this fracturing mechanism. (auth)

NSA, vol. 26, 1972

14531 (AI-AEC-13085) BURNABLE POISON DEPOSITION ON ZIRCONIUM HYDRIDE FUEL: PROCESS DEVELOPMENT AND EVALUATION. Van Houten, R. (Atomics International Div., Canoga Park, Calif. (USA)). 22 Jun 1973. Contract AT-(04-3)-701. 62p. Dep. NTIS $5.25.

The development and evaluation of a process for depositing thin rare earth oxide burnable poison films on fuel rods are reviewed. The burnable poison films are formed by controlled-rate, electron-beam-heated, high-vacuum evaporation of the oxides from water-cooled crucibles. Coating uniformity (mg poison/cm^2) and reproducibility are good (to ±5%), and rapid, accurate nondestructive x-ray fluorescence analytical procedures permit up to 100% inspection (all surface areas of all coated fuel rods). Poison placement screening criteria, recommended coating and measuring equipment performance, and swelling and aging test results are given. All test results for Gd_2O_3, Er_2O_3, and Y_2O_3 coatings are promising. Vapor-deposited coatings containing Eu_2O_3 and/or Sm_2O_3 are not recommended, because of rapid transfer of Eu and Sm to the SCB hydrogen barrier; but preliminary tests on the incorporation of up to 9 wt % Eu in an SCB-type sleeve showed good Eu retention through 1510 hr at 1430°F, and good SCB stability. Tests extended to 18,584 hr at 1300°F, to 10,000 hr at 1400°F, and to 5000 hr at 1500°F. (auth)

NSA, vol. 28, 1973

FUEL ELEMENT FAILURES CAUSED BY IODINE STRESS CORROSION

KETIL VIDEM and LIV LUNDE
Institutt for Atomenergi, 2007 Kjeller, Norway

Abstract—Sections of unirradiated cladding tubes were plugged in both ends by mechanical seals and internally pressurized with argon containing iodine.

The time to failure and the strain at failure as a function of stress was determined for tubing with different heat treatments. Fully annealed tubes suffer cracking at the lowest stress but exhibit the largest strains at failure.

Elementary iodine is not required. Small amounts of iodides of zirconium, iron and aluminium can also give cracking. Moisture, however, was found to act as an inhibitor.

A deformation threshold exists below which stress corrosion failure does not occur regardless of the exposure time. This deformation limit is lower the harder the tube. The deformation at failure is dependent on the deformation rate and has a minimum at 0.1%/hr. At higher deformation rates the failure deformation increases, but only slightly for hard tubes.

Fuel was over power tested at ramp rates varying between 0.26 to 30 W/cm min. For one series of fuel pins the failure deformations of 0.8% at high ramp rates were in good agreement with predictions based on stress corrosion experiments. For another series of experiments the failure deformation was surprisingly low, about 0.2%.

ANNALS NUC. EN., vol. 3, 1976

3. The Performance of Boron-Containing Control Rods in Water-Cooled Power Reactors/*H. E. Williamson (GE-San Jose), Invited Paper*

Boron alloyed with stainless steel and bulk boron carbide clad in stainless steel are the forms in which boron has been used in water-cooled power reactor control rods. Boron stainless-steel alloys suffer severe irradiation embrittlement, and a high incidence of cracking in the vicinity of welds has been observed after reactor service. Small-diameter core-length stainless-steel tubes filled with boron carbide powder and arranged in crossing rows to form a cruciform cross section have proved to be a reliable and economical[1] control rod. The application of control rods of this type in large power reactors was pioneered in the Dresden Nuclear Power Station[2]. Control rods of this design were installed in the Dresden Nuclear Power Station in 1961 and continue to operate satisfactorily. At this time there are about 270 control rods of this type in six operating boiling-water power reactors.

Prior to the 1965 Dresden refueling outage, the 80 Dresden B_4C control rods had been in service for about four years. No measurable loss in control strength as a result of their operation has been detected. Control-rod-blade examination by an underwater television camera, during this refueling outage, showed no deterioration or distortion of the control blade had taken place. During this refueling outage a programmed exchanged of some high- and low-duty control rods was performed. One set of B_4C control blades, properly programmed, may well last for 20 calendar years of reactor operation at 0.8 capacity factor[2].

Concurrent with the installation of the boron-carbide-in-tube control rods in Dresden, boron-carbide filled stainless-steel tube samples representative of the tubes in the control blades were inserted in the instrument tubes of two fuel assemblies. Examinations of these sample tubes provide experimental information on the effects of irradiation on the boron-carbide-filled tubes well in advance of exposures achieved in the control blades themselves. The sample tubes remain in the reactor core continuously as contrasted to the programmed operation of control rods.

Sample tubes have been examined, one after 2925 h of irradiation (3.4% B-10 burnup) and one after 11 040 h of irradiation (8.5% B-10 burnup). Data on the B_4C powder performance, such as helium release, has been reported previously[3]. Subsequently, additional measurements have been made on the performance of the B_4C-filled sample tubes. A sample rod which is a duplicate of the B_4C-filled tubes that were used in control-rod blades was examined visually and with ultrasonic techniques during the 1965 Dresden refueling outage (≈ 26 000 h of turbine-generator service since insertion) and returned to the core for continued irradiation. The ultrasonic testing indicated the cladding tube containing the B_4C powder column was free of defects.

(Cont'd pg 74)

Some of the sample rods were equipped with a device to measure swelling or axial shrinkage (further compaction) of the 65- to 75%-dense B_4C powder column. One of these was examined after 2925 h of operation and one after 11 040 h of operation. The first indicated 0.23 in. (0.7%) of shrinkage and the second indicated 0.39 in. (1.1%) of shrinkage for an approximately 35-in.-long column of B_4C. Metallographic sectioning of the rods confirmed that the devices had satisfactorily followed the B_4C column. This small amount of shrinkage is of no consequence to the performance of the control rods in which the B_4C columns are compartmented by a series of stainless-steel balls in the tubes.

Previous measurements on helium gas release[2] from the B_4C powder in the sample tubes after irradiation had been made by evacuation at room temperature, and were subject to the uncertainty that some helium might have remained adsorbed on the B_4C powder at room temperature. Subsequent annealing of the powder in vacuum at temperatures up to 750°F shows that essentially all (≈ 95%) the helium released during irradiation was extracted by vacuum at room temperature. The temperature of the B_4C powder during irradiation was less than the 750°F annealing temperature.

The performance of the B_4C-filled stainless-steel-tube control rods continues to be satisfactory. The effects of irradiation on the B_4C powder has been well within performance characteristics used in the control rod design. Continued surveillance planned for individual B_4C sample tubes and on in-service performance of blades now operating in boiling water reactors is expected to lead to continuing improvements in the design of control rod blades for General Electric reactors.

1. WILLIAMSON, H. E., and F. H. MEGERTH, "Economic Evaluation of Control Rod Materials and Fabrication Processes," GEAP-4013, Informal AEC Research and Development Report (May, 1962).
2. BRAMMER, H. A., and J. JACOBSON, "Design, Fabrication and Performance of Boron Carbide Control Elements," APED-4370, presented November, 1963.
3. WILLIAMSON, H. E., "The Performance of B_4C Powder in the Dresden Control Rods," *Trans. Am. Nucl. Soc.*, 7, 1, 98 (1964).

ANS TRANS., vol. 9, 1966

14641 METHOD OF FABRICATING A NUCLEAR REACTOR FUEL ELEMENT. Woolsey, C. C. Jr. (to United States Atomic Energy Commission). US Patent 3,809,731. 7 May 1974. Filed date 31 Mar 1961. 4p.

An improved method of fabricating uranium–zirconium hydride fuel element and a method for protecting the fuel element from interaction with the cladding material at high temperatures con-

sists of preparing a Zr–U alloy, enclosing the alloy in a sleeve of Zr metal and then hydriding to yield a U–Zr alloy hydride fueled core clad with a protective zirconium hydride outer region. The resulting fuel rod is then enclosed in a conventional protective tubing such as stainless steel. This provides a barrier material between the fuel-moderator material and the cladding and prevents eutectic formation with the U fuel. (FS)

NSA, vol. 30, 1974

6. Environmentally Influenced Failure of UO_2-Zircaloy Fuel, *J. C. Wood (AECL-CRNL), invited*

Zircaloy cladding of UO_2 fuel sometimes cracks when fuel power is increased.[1] Defect probability in these ramping failures increases with power, power increase, burnup, and dwell time at high power.[2] Possible defect mechanisms are stress corrosion cracking (SCC) by fission products such as iodine[3] or local mechanical overload of Zircaloy at UO_2 cracks. The purpose of this paper is to show the many similarities between ramping defects and SCC.

Iodine-induced SCC will occur if:

1. The Zircaloy is in a metallurgically susceptible state—a rapid increase in susceptibility of fuel cladding occurs at a neutron dose $\approx 10^{20}$ n·cm^{-2} (E > 1 MeV).

2. There is enough corrodant—vapor pressures of iodine $<10^{-4}$ MN/m^2 can cause SCC, but for failures within 1 h at temperatures and stresses (caused by UO_2 expansion) pertinent to operating conditions, an iodine pressure of 10^{-2} MN/m^2 was required.

3. There is enough sustained stress—the critical stress for SCC depends on exposure time: yield stress (or higher) is required for failure within 1 h, but at t > 100 h less than half the yield stress is required to cause failures.[4]

SCC can occur after surface ZrO_2 is cracked, by straining the Zircaloy substrate, or is penetrated by a chemical (pitting) attack at long exposure times; if plastic straining is a prerequisite for SCC, the average circumferential strain required is only ~0.1%.

Most power ramping defects are characterized by a time delay between power increase and defect signal. The time, t, dependencies of ramping defects and laboratory SCC tests are of the form $1 - \exp(-\lambda t)$, where $\lambda = 2.3$ h^{-1} and 1.8 h^{-1}, respectively. The defects are branching, narrow cracks that occur at circumferential ridges and over UO_2 cracks, where stress and strain are concentrated. The Zircaloy cracks have flat pseudo-cleavage facets on the fracture faces, observable by SEM, that cannot be reproduced in the absence of a corrodant; the facets are indistinguishable from those produced by SCC in iodine.

Fuel elements comprising irradiated cladding and fresh UO_2 (no fission products) have been ramped to high powers without causing defects.[5] When iodine was introduced deliberately, defects occurred but not if iodine and water were included together: laboratory tests also show an inhibitive effect of moisture on SCC. The influence of environment was also underlined when a UO_2 chip was lodged in the fuel-cladding gap: the cladding crack started from the inside surface although maximum tensile stress was at the outside.[6]

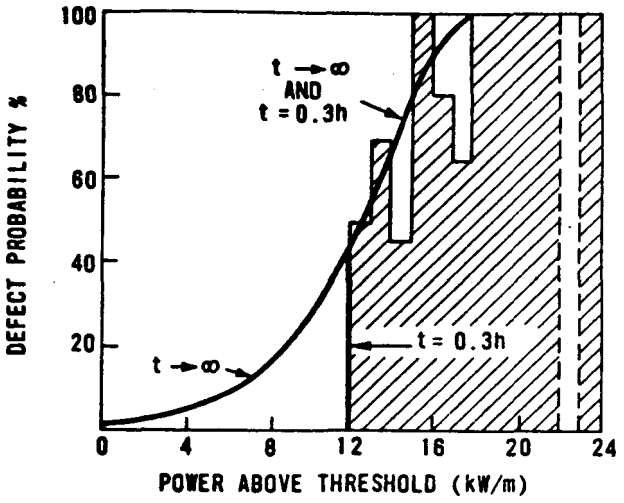


Fig. 1. Defect probability as a function of fuel power at short and long dwell times; a sudden step also occurs in probability (at 30%) if the abscissa is power increase and dwell time is 0.3 h.

(Cont'd pg 76)

CANDU reactors use on-power-fueling, which involves moving fuel, previously irradiated at low power, to high power for short periods; e.g., Pickering = 0.3 h; and then moving to more moderate power.2 Defect probability is shown in Fig. 1 as a function of power above the threshold to cause defects. The smooth curve refers to fuel that resided for long periods at high power. The data for t = 0.3 h (histogram, Fig. 1) show a sudden cutoff. This is consistent with SCC being the failure mechanism: there is only time for ZrO2 cracking to occur and not the slow penetration of the ZrO2 layer by chemical/mechanical processes, so the distribution that would have resulted for large t was chopped.

The new results included here are;

1. effects of neutron dose
2. low circumferential strains
3. fractography at various strain rates and temperatures without iodine
4. time dependencies
5. iodine-doped fuel pins with and without water present
6. the chopping of defect probability distribution at t - 0.3 h.

These points corroborate our previous conclusion that most ramping defects are caused by SCC.1 2 3 4 5 6

1. E. ROLSTAD and K. D. KNUDSEN, *Proc. Fourth Geneva Conf.,* 10, 257 (1971).
2. W. J. PENN, R. K. LO, and J. C. WOOD, *Trans. Am. Nucl. Soc.,* 22, 209 (1975).
3. H. S. ROSENBAUM, *Electrochem. Tecknol.,* 4, 153 (1966).
4. C. C. BUSBY, R. P. TUCKER, and J. E. McCAULEY, *J. Nucl. Mater.,* 55, 64 (1975).
5. R. D. MacDONALD, D. G. HARDY, and C. E. L. HUNT, *Trans. Am. Nucl. Soc.,* 17, 216 (1973).
6. A. S. BAIN, J. C. WOOD, and C. E. COLEMAN, BNES Conference paper 56 (Oct. 1973).

ANS TRANS., vol. 23, 1976

9171 (ANL-7801) SEMIAUTOMATIC APPARATUS FOR CREEP AND STRESS-RUPTURE TESTS OF THIN-WALL FUEL-CLADDING TUBES UNDER INTERNAL GAS-PRESSURE LOADING. Yaggee, F. L.; Styles, J. W.; Brak, S. B. (Argonne National Lab., 111.). Sep 1971. Contract W-31-109-eng-38. 26p. **Dep.** NTIS.

A semiautomatic apparatus is described for creep and stress-rupture tests of small-diameter, thin-wall tubes under biaxial load. Twenty-four specimens, arranged in four bundles each with six specimens, are accommodated in a single test. Each bundle is connected to a common source of high-pressure gas. Creep or stress-rupture tests can be conducted under constant-load (constant-pressure) or constant-stress (decreasing-pressure) conditions at temperatures to 1200°C in a vacuum or an inert-gas environment. Biaxial creep tests were conducted on Types 304 and 304L stainless steel tubes at temperatures between 550 and TOOX and applied tangential stresses between 7500 and 40,000 psi. Test results have yielded a value of 6.7 for the stress dependence of the steady-state creep rate and 95,000 cal/mole for the activation energy of creep. Both values are in good agreement with the published values for 18-8 austenitic stainless steel determined in uniaxial creep tests. Experimental results provide a strain profile of the three types of specimen failures normally observed: catastrophic rupture, pinhole leak, and fissure. The results also yield information on the change in material density as a function of diametral strain, .strain rate, and temperature, (auth)

NSA, vol. 26, 1972

7169 (CONF-741232-, pp 127-130) ANALYSIS AND VERIFICATION OF CREEP COLLAPSE OFZIRCALOY-4 FUEL CLADDING. Yoon, K. E. (Babcock and Wilcox Co., Lynchburg, VA). 1974.

From European conference on irradiation behaviour of fuel cladding and core component materials; Karlsruhe, F. R. Germany (3 Dec 1974).

In European conference on irradiation behaviour of fuel cladding and core component materials.

An analytical method based on W. K. Wilson's method for creep-buckling of tubes under external pressure (WAPD-TM-956), is presented. This method and the CROV computer code is verified against experimental creep collapse data for Fr-4 cladding at 343 and 371*C both in- and ex-reactor at three pressure differentials.

NSA, vol. 32, 1975

3. Analysis of Fuel Cladding Collapse Due to Creep, *K. E. Yoon, A. F. J. Eckert (B&W)*

Recent concern over fuel densification created renewed interest in nuclear fuel cladding collapse analysis under LWR operating conditions. This has led to development of a more accurate and reliable analytical method on this subject.

A number of papers were published in this area in the past but none was compared with any experimental result, simply because of the lack of such long-term collapse data. Another difficulty in this analysis is the creep behavior of the fuel cladding material itself. It is generally known that Zircaloy cladding creep properties are dependent on stress, temperature, flux level, and manufacturing processes. Accurate characterization of the primary and secondary creep properties for a particular cladding type is a costly and difficult task.

Babcock and Wilcox Company developed an improved analytical method based on a theory similar to that of Wilson.[1] The anisotropic properties of the cladding and refined method of incremental creep deformation were included in the analysis.

An Ibrahim-type equation for the thermal creep rate expression was used with empirical constants determined from an extensive internally pressurized creep test conducted in B&W. Test procedures and results are presented in Ref. 2. The irradiation creep rate was adopted to be proportional to the flux times the generalized stress from the literature. The anisotropy constants which were used in the generalized stress expression and the flow rule in this analysis were obtained from a short-time biaxial plastic test. These material properties were incorporated in the analytical procedure which was developed into the computer program CROV. To prove the validity of this analysis, completely independent fuel cladding creep collapse experiments were conducted using specimens made by the same process as those used in the creep tests. This is discussed by Papazoglou.[3] Comparison of predicted and observed ovality-time behavior is shown in Fig. 1. The analytical prediction is always conservative. This is due to the following facts: (a) a strain hardening creep law was used which is conservative for the gradually increasing stress state on the apogee of the cladding; (b) the creep rate is fixed after the generalized strain reaches a certain criterion which precludes any continuing strain hardening effects, and (c) the creep analysis assumes a "worst case" of an initially uniform oval shape. Nonuniformities are shown to delay the development of ovality and subsequent collapse.[3]

Based on accurate creep and mechanical property characterization and good correlation with the independent creep collapse experimental results, this method of analysis is shown to be a reliable tool for predicting creep collapse behavior in a conservative manner.

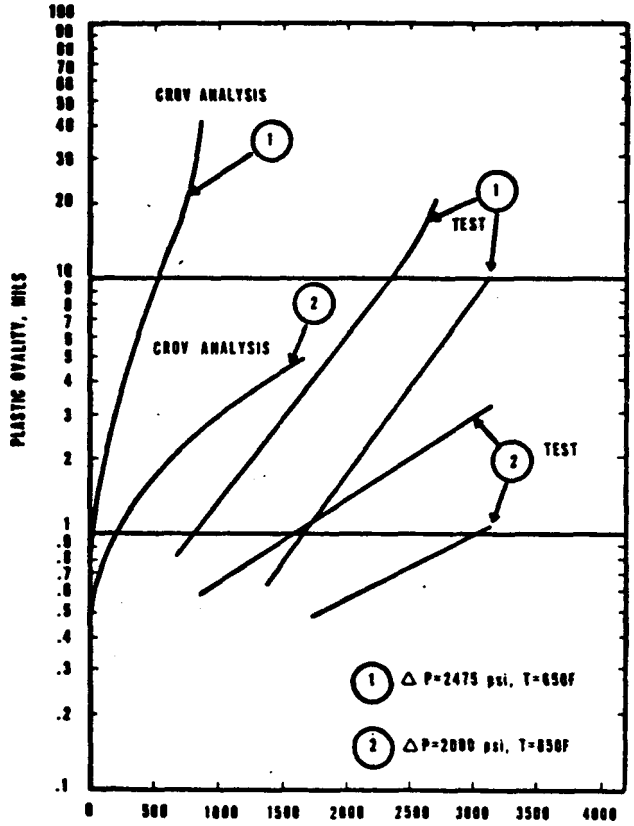


Fig. 1. Analysis vs test results ovality-time behavior.

1. W. K. WILSON, "Method of Analysis for the Creep-Buckling of Tubes Under External Pressure," WAPD-TM-956, Westinghouse Electric Corp (Oct. 1970).
2. G. S. CLEVINGER, "Ex-Reactor Biaxial Creep of Zircaloy-4 Cladding," *Trans. Am. Nucl. Soc.*, 19, 143 (1974).
3. T. P. PAPAZOGLOU, "Creep Collapse of Zircaloy-4 Cladding," *Trans. Am. Nucl. Soc.*, 19, 143 (1974).

Blank Page

III. EFFECT OF FUEL MICROSTRUCTURE ON PERFORMANCE AND FUEL DENSIFICATION

Factors Affecting Fuel Densification

Blank Page

M163 (CENPD-118(Rev.1)) DENSIFICATION OF COMBUSTION ENGINEERING FUEL. Andrews, M. G. (comp.). (Combustion Engineering, Inc., Windsor, Conn. (USA)). May 1974. 22p.

Experimental fuel densification information obtained by C-E from irradiations at Yankee Rowe, the Battelle Research Reactor, Palisades, and the Halden Test Reactor was used to establish the parameters important to the production of dimensionally stable fuels. The experimental results are sufficient to demonstrate that the manufacturing modifications applied by C-E have been effective in reducing the extent of densification. The densification kinetics data obtained from the study were used to supplement existing results from which an empirical expression for the dependence of densification on burnup could be derived. In addition, archive pellets from various experimental batches of pellets were used to establish a thermal resintering test as a predictor of in-pile densification. A summary of the most important results of the study is given. (auth)

NSA, vol. 30, 1974

29174 (TID-26458) TECHNICAL REPORT ON DENSIFICATION OF EXXON NUCLEAR BWR FUELS. (Atomic Energy Commission, Washington, D. C.). 4 Sep 1973. 34p. Dep. NTIS $3.75.

Fuel densification (gaps in fuel pellet column), its mechanism, and its effects are discussed. Models used by Exxon to predict the consequences of fuel densification are given, and the analysis required for determining the effects on reactor operation and safety is described. Findings and conclusions of the Regulatory Staff review and evaluation of the Exxon BWR fuel (Oyster Creek) are given. (DLC)

NSA, vol. 28, 1973

31796 (TID-26459) TECHNICAL REPORT ON DENSIFICATION OF GENERAL ELECTRIC REACTOR FUELS. (Atomic Energy Commission, Washington, D. C.). 23 Aug 1973. 73p. Dep. NTIS $5.75.

Fuel densification mechanisms and effects; the models used to predict the consequences of fuel densification; and essential elements of analyses required to determine the effects of fuel densification on reactor normal operating modes, anticipated transients, and possible accident sequences are presented. It is concluded that irradiation induced densification occurs to 96.5% of theoretical density (geometric) in BWR fuels and when the reactor is operated at substantial power, densification is instantaneous. Fuel densification causes the fuel rod to contain stored energy, increases the linear heat generation rate, decreases heat transfer between the fuel pellet and cladding, and creates the potential for a power spike in the collapsed section of the fuel rod. (FS)

NSA, vol. 31, 1975

23162 SOME OBSERVATIONS OF DENSITY AND POROSITY CHANGES IN UO_2 FUEL IRRADIATED IN WATER-COOLED REACTORS. Banks, D. A. (UKAEA). J. Nucl. Mater.; 54: No. 1, 97-107(Nov 1974).

A technique was developed for the hot-cell measurement of the apparent density of irradiated UO_2 fuel extraction from a fuel pin. A single determination is accurate to ±3% at the 95% confidence limit. The method was applied to fuel irradiated in thermal neutron fluxes in the Winfrith SGHWR and in the Halden BWR. Material was examined at ratings of 1 to 51 W/g and in the burn-up range $0.09 - 5.79 \times 10^{20}$ fissions/cm^3. It is concluded that pellets with peak temperatures below 1100°C densify during irradiation, but at higher temperatures the pellets begin to swell. Fuel micrography shows that the densification is principally due to the loss of micropores with a temperature dependency given by an activation energy of 5200 cal/mol. Above 1000°C the densification is masked by the formation and growth of intergranular fission gas bubbles, whose volume may exceed that of the manufactured pores which have sintered. In solid fuel pellets central swelling did not balance densification in the cooler rim until the fuel center temperature exceeded 1700°C. (auth)

23163 HEAT TRANSFER AND THERMAL CONDUCTIVITY OF MULTI-HUNDRED WATT $^{238}PuO_2$ FUEL SPHERES. Tennery, V. J. (Mound Lab., Miamisburg, Ohio (USA)). J. Nucl. Mater.; 54: No. 1, 73-78(Nov 1974).

NSA, vol. 31, 1975

38951 REDISTRIBUTION OF PLUTONIUM AND URANIUM IN MIXED (U,Pu) OXIDE FUEL MATERIALS IN A THERMAL GRADIENT. Bober, M.; Sari, C.; Schumacher, G. (Kernforschungszentrum, Karlsruhe, Ger.). J. Nucl. Mater.; 39: No. 3, 265-84(Jun 1971).

The redistribution of plutonium and uranium has been investigated in mixed ($U_{0.85}Pu_{0.15}$) oxide fuel materials subjected to a temperature gradient in laboratory experiments. The cylindrical specimens in sealed tungsten containers were exposed to temperatures up to 2,800°K and longitudinal temperature gradients up to 2,000°K/cm. The structures which developed in the specimens were correlated exactly with temperature. An inspection by alpha-counting of the specimens after thermal treatment showed a decrease of plutonium in the columnar grain region and an accumulation at the high-temperature end. Three redistribution processes were indicated and are discussed in detail. Each of them caused a characteristic distribution of plutonium concentration. The processes were: preferential evaporation at the surface of the specimens; preferential evaporation and condensation in migrating voids; and thermal diffusion. For thermal diffusion a heat of transfer of 35 kcal/mole was estimated and long time effects were calculated. In conclusion, corresponding effects of the redistribution processes in reactor fuel elements are discussed briefly. (auth)

NSA, vol. 26, 1972

JOURNAL OF APPLIED PHYSICS 1956 VOLUME 27, NUMBER 11 NOVEMBER, 1

Phase Changes in Pile-Irradiated Uranium-Base Alloys

M. L. BLEIBERG, L. J. JONES, AND B. LUSTMAN
Bettis Plant, Westinghouse Electric Corporation, Pittsburgh 30, Pennsylvania

(Received May 9, 1956)

Specimens of U-9, -10.5, -12, and -13.5 w/o (weight percent) Mo and U-10 w/o Nb, heat treated to retain the metastable gamma phase at room temperature and also to transform this phase to the room temperature stable phases, were irradiated in the Materials Testing Reactor (MTR) at maximum temperatures of less than 200°C. Electrical resistivity, temperature coefficient of electrical resistivity, hardness, and density measurements as well as x-ray diffraction patterns indicated that the stable phases at room temperature reverted to the metastable gamma phase during irradiation. No apparent changes in the microstructures of thermally transformed specimens were observed after irradiation., Lack of observable microstructural changes was attributed to the development of a ghost structure. X-ray line broadening measurements indicated essentially complete homogenization of thermally transformed U-Mo alloy specimens during irradiation. The mechanism of the phase reversal was explained on the basis of the displacement spike hypothesis.

1. Densification Kinetics in UO_2 Fuels, *B. J. Buescher, G. R. Horn (B&W)*

A model has been developed to predict the kinetics of in-reactor densification. The predictive technique uses out-of-reactor resintering behavior to determine values for parameters to be used in Marlowe's model of in-reactor densification.[1]

$$\rho = \rho_0 \exp(-S\dot{F}t) + \frac{M}{A} \exp\left[-S\left(\frac{G_0^3}{AD^0_{irr}} + \dot{F}t\right)\right] \times \ln\left(1 + \frac{AD^0_{irr}}{G_0^3}\dot{F}t\right) , \quad (1)$$

where

ρ = density after irradiation

ρ_0 = initial density

S = volumetric swelling rate for 100% dense fuel

$\dot{F}$ = fission rate

t = irradiation time

M = densification rate constant

A = grain growth rate constant

G_0 = initial grain size

D^0_{irr} = constant relating irradiation-induced diffusivity to fission rate.

The major difficulty with Marlowe's model is the fact that precise values for the parameters M, A, D^0_{irr}, and G_0 are very difficult to determine experimentally and (except for D^0_{irr}) can vary significantly. This difficulty can be partially overcome by defining a new parameter

$$B_0 = \frac{AD^0_{irr}}{G_0^3} . \quad (2)$$

The problem then becomes one of determining values for the two parameters, M/A and B_0.

Marlowe's model is an extension of Coble's treatment of intermediate and final-stage thermal sintering.[2] Coble derived the following equation for diffusion-controlled thermal sintering:

$$\rho - \rho_0 = \frac{M}{A} \ln\left(1 + \frac{ADt}{G_0^3}\right) , \quad (3)$$

where ρ is the density after resintering, D is the thermally activated bulk diffusion rate, and t is the resintering time. The parameters M, A, and G_0 are the same in both Eqs. (1) and (3).

Again, it is convenient to define a new parameter

$$B = \frac{AD}{G_0^3} . \quad (4)$$

Coble's equation then becomes

$$\Delta\rho = \frac{M}{A} \ln(1 + Bt) \quad (5)$$

with two unknowns, M/A and B. By conducting resintering tests at two different times, two equations in two unknowns can be written and solved simultaneously to obtain values for M/A and B. Comparing Eqs. (2) and (4), it is apparent that

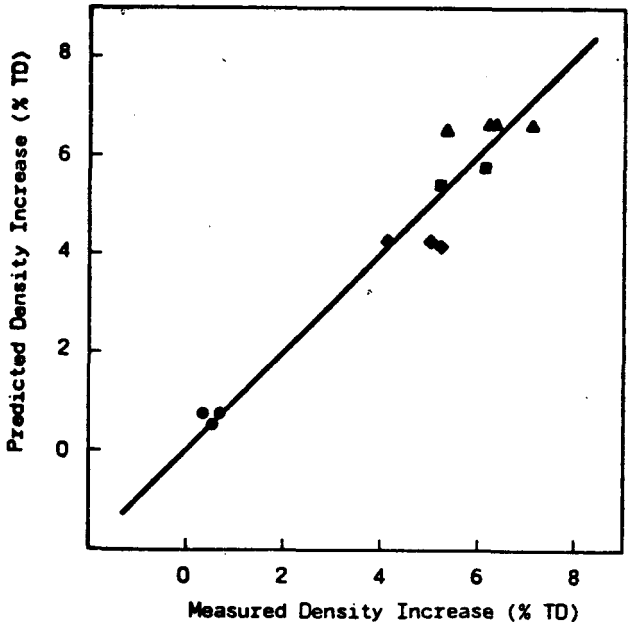


Fig. 1. Predicted versus measured density increases of irradiated fuel pellets.

$$B_0 = B\frac{D^0_{irr}}{D} . \quad (6)$$

Once the ratio of the fission-induced diffusion rate to the thermal diffusion rate has been established, values for the two unknowns, M/A and B_0, can be determined for predicting the kinetics of in-reactor densification.

The ratio of the irradiation-induced diffusivity to the thermal diffusion rate has been determined using the data generated in the EEI/EPRI-sponsored "UO_2 Densification Program"[3,4] for pellet types 1, 2, 3, and 4. Values for M/A and B were determined for each of these pellet types by a least-squares fit of Eq. (5) to the density increases measured after resintering these fuel types at 1700°C for 4, 14, and 48 h. With these values of M/A and B and the in-reactor densification data generated in the EEI program for types 1, 2, 3, and 4, a nonlinear regression analysis was done using Eqs. (1) and (6) to determine the ratio of D^0_{irr}/D. Since there is an apparent power threshold, the analysis was made using the data from fuel pellets operated in excess of 5 kW/ft. The ratio for D^0_{irr}/D was found to be

$$D^0_{irr}/D = 1.15 \times 10^{-15} \text{ cm}^3 \text{ sec/fission} . \quad (7)$$

(Cont'd pg 83)

The predicted versus measured in-reactor density increases are shown in Fig. 1. In this figure, only those pellets irradiated at average linear heat rates >5 kW/ft are shown. As can be seen, the density increases predicted for fuel pellets irradiated at linear heat rates above 5 kW/ft are in excellent agreement with the observed values and this gives strong support to the validity of the model. For fuel irradiated at linear heat rates <5 kW/ft, the correlation overpredicts the data.

1. M. O. MARLOWE, "In-Reactor Densification Behavior of UO_2," NEDO-12440 (July 1973).

2. R. L. COBLE, "Sintering Crystalline Solids-I Intermediate and Final State Diffusion Models," *J. Applied Phys.*, 32, 787 (1961).

3. M. D. FRESHLEY et al., "The Effect of Pellet Characteristics and Irradiation Conditions on UO_2 Fuel Densification," Reported in Proceeding of Joint Topical Meeting on Commercial Nuclear Fuel Technology Today, p. 2-106 (1975).

4. "EEI/EPRI Fuel Densification Report," EPRI 131 (Mar. 1975); prepared by Battelle, Pacific Northwest Laboratories.

ANS TRANS., vol. 22, 1975

13048 CREEP AND DENSIFICATION OF UO_2 UNDER IRRADIATION. Brucklacher, D.; Dienst, W. (Kernforschungszentrum, Karlsruhe, Ger.). pp 60.1-60.3 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004-.

Fuel dimensional changes can result from fuel swelling and from densification by sintering under irradiation. Swelling could be accomodated by irradiation-induced creep under cladding restraint. Fuel swelling rates were measured on UO_2 at low temperatures. Data on UO_2 sintering under irradiation were obtained by various measurements and are compared with out-of-pile results. At low temperatures sintering is enhanced by irradiation up to 1300°C. But UO_2 pellets of an original density above 96% TD will not show any considerable densification. In-pile creep experiments on UO_2 were performed at fuel temperatures between 250 and 850°C. The creep rates are proportional to stress and also to fission rate and are only slightly dependent on temperature. Results are discussed with regard to mechanical interaction between fuel and cladding. 5 references. (auth)

NSA, vol. 32, 1975

16636 FUEL PERFORMANCE: IMPACT OF DENSIFICATION. Chernock, W. P.; Duncan, R. N.; Harkness, S. D. (Combustion Engineering, Inc., Windsor, CT). Proc. Am. Power Conf.; 36: 234-240(1974).

From American power conference; Chicago (29 Apr 1974). See CONF-7404100-.

Results of in-reactor tests designed to study the extent and kinetics of densification for fuels with various microstructures and initial densities are presented. A review of mechanisms associated with in-reactor densification is included to provide a link between theory and the experimental observations showing the dependence of densification on UO_2 microstructures. (11 references) (DCC)

NSA, vol. 32, 1975

18179 CREEP BEHAVIOR OF CERAMIC NUCLEAR FUELS UNDER NEUTRON IRRADIATION. Brucklacher, D.; Dienst, W. (Kernforschungszentrum, Karlsruhe, Ger.). J. Nucl. Mater.; 42: No. 3, 285-96(Mar 1972).

A theoretical estimation of the irradiation-induced creep rate of UO_2 resulted in a creep rate range between about 6×10^{-6}/h and 8×10^{-5}/h for a fission rate of 1×10^{14} f/cm^3·s and a stress of 2 kgf/mm^2. It is essentially due to the "thermal rods" along the fission fragment tracks. Therefore, creep rates should only weakly depend on temperature (below 1000 to 1200°C) and must be markedly lower for carbide and nitride fuel. To prove this estimate by experiments, several types of in-pile creep capsules were developed by which the length or diameter change of pneumatically loaded fuel samples can be measured continuously. In-pile creep experiments on UO_2 were performed at fuel temperatures between 250 and 850°C. The creep rates were proportional to stress (0 to 4 kgf/mm^2) and, in a preliminary approximation, also to fission rate (1×10^{13} to 2×10^{14} f/cm^3·s) and were in the range estimated before. Within the limits of accuracy, the in-pile creep rate was independent of temperature (250 to 850°C). In-pile creep tests on some UN samples resulted in creep rates that are lower by an order of magnitude than for UO_2 under comparable conditions. The porosity dependence of the UN in-pile creep rate was found to agree with an adequate out-of-pile relationship for ceramic materials. (auth)

NSA, vol. 26, 1972

36618 (HW-67110) IRRADIATION OF NATURAL URANIUM SEVEN-ROD CLUSTER FUEL ELEMENTS IN THE ENGINEERING TEST REACTOR, GEH-10-2 (RM-281). Final Report. Call, R. L.; Claudson, T. T.; McMahan, M. E. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Mar 1962. Contract AT(45-1)-1350. 33p. Dep. NTIS.

Declassified 4 Nov 1971.

Three seven-rod cluster, natural uranium fuel elements fabricated from coextruded Zircaloy-2-clad uranium were irradiated to an average exposure of 950 MWd/t. The core center temperature of the elements during the major part of the irradiation was 405°C with a specific power of 67.5 kW/ft. During the latter part of the irradiation core temperature was 516°C with a specific power of 125.5 kW/ft. Except for the failure of the end supports of two of the clusters during or after discharge, the general appearance of the clusters was good. Some of the rods were warped slightly but their surfaces were smooth and free of bumps. No film was deposited on the rods and no evidence of cladding or weld corrosion was observed. The rod which was measured had increased in volume 1.64 percent at 0.18 percent burnup of its uranium atoms. There were no cracks in the uranium and the uranium-to-cladding bond layer was sound. The uranium microstructure was characteristic of irradiated uranium. Results of the test show that seven-rod cluster fuel elements can be irradiated successfully at high temperatures and powers to exposures as high as 1000 MWd/t. The failure of the end supports was not caused by irradiation, per se, but was a result of the need to remove material from the end supports so that the fuel elements could be fitted into the basket for irradiation. (auth)

NSA, vol. 26, 1972

41320 (HEDL-TME-71-80) MICROSTRUCTURE CHARACTERISTICS OF PNL-1 AND 2 IRRADIATION TEST FUELS. **Carlson, M. C. J.** (Hanford Engineering Development Lab., Richland, Wash.). **May 1971.** Contract AT(45-1)-2170. 27p. Dep. NTIS.

Measurements providing quantitative porosity characterization, grain structure characterization and composition distribution evaluation for representative fuel batches used in EBR-II irradiation testing subassemblies PNL-1 and PNL-2 are described. Pore size-frequency distributions are shown for fuel batches MEE-9, MEE-12 and MEE-13 over the size range 0.2 to 128 microns. The results of submicron porosity evaluation from a cathodic vacuum etched MEE-9 sample indicate this technique is a promising approach to porosity observation and measurement to sizes as small as 300 Å. Pore size-frequency distribution and volume for pores over the size range 300 to 800 Å are shown for as-fabricated MEE-9 fuel. The relationship of porosity characteristics observed from the cathodic vacuum etched MEE-9 pellet surface to the characteristics observed from as-polished and acid etched surfaces is shown. Comparison of porosity evaluation by replica and direct surface observations in the scanning electron microscope shows no significant difference between the results from the two methods. The homogeneity Figure of Merit for PNL-1 and 2 fuel batch MEE-12 was calculated to be 0.999 as compared with 1.00 for a completely homogeneous material. Result of analysis showed inclusions containing stainless steel, aluminum, and tungsten. (auth)

NSA, **vol.** 26, 1972

15572 (HEDL-SA-245) QUANTITATIVE MICROSTRUCTURE AND COMPOSITION ANALYSIS. Carlson, M. C. J. (Hanford Engineering Development Lab., Richland, Wash.). 19 Oct 1971. Contract AT(45-1)-2170. 32p. (CONF-711030-2). Dep. NTIS.

From twenty-fourth Pacific Coast regional meeting of the American Ceramic Society; Anaheim, Calif. (30 Oct 1971).

A description of methods used in measuring radiation effects on fuels is presented. Information is included on measurement scope, porosity characterization, pore frequencies and volume, surface porosity, relations of porosity with physical properties, grain characterization, and composition distributions. (J.R.D.)

NSA, vol. 26, 1972

13643 BEHAVIOR OF ZIRCALOY-UO_2 FUEL ELEMENTS WORKING AT LOW TEMPERATURE (300°C). Chagrot, M.; Ringot, C.; Vidal, H. (CEA, Saclay, France). pp 91.1-91.4 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004.

Some irradiation tests have been performed to verify the behaviour of an "applied" sheath in UO_2 - zircaloy fuel element, when there is an interaction between the fuel and the sheath due, for example, to a power increase. The EI.O tests had three purposes: determine the failure mechanism, in view of making possible counter measures, select the more favourable cladding material, measure the strains resulting from power increases during normal reactor operation. The main results have shown that several rupture mechanisms exist, one being purely mechanical, and that it seems that a cladding having a high initial ductility has a better behaviour. (auth)

NSA, vol. 32, 1975

257799 Chaudhry, S.M.; Anand, A.K. (Bhabha Atomic Research Centre, Bombay (India). Reactor Engineering Div.). Bhabha Atomic Research Centre, Bombay (India). **Creep collapse of TAPS fuel cladding. INIS-mf-3108/29.** 1975. 7 p. 9 refs. Symposium on structural mechanics, Bombay, India, 20 Mar 1975.

Densification of UO_2 can cause axial gaps between fuel pelets and cladding in unsupported (internally) at these regions. An analysis is carried out regarding the possibility of creep collapse in these regions. The analysis is based on Timoshenko's theory of collapse. At various times during the residence of fuel in reactor following parameters are calculated : (1) inelastic collapse of perfectly circular tubes (2) plastic instability in oval tubes (3) effect of creep on ovality. Creep is considered to be a non-linear combination of the following : (a) thermal creep (b) intresenic creep (c) stress aided radiation enhanced (d) stress free growth (4) Critical pressure ratio. The results obtained are compared with G.E. predictions. The results do not predict collapse of TAPS fuel cladding for five year residence time. (author).

FUEL CANS: failures.

Atomindex, vol. 7, 1976

In-Pile Densification of Uranium Dioxide, *W. P. Chernock, M. G. Andrews, S. D. Harkness (Combustion Engng–USA)*

In-reactor densification of UO_2 fuels has been identified in power reactor fuels.[1] Results of extensive in-reactor testing of UO_2 fuel by Combustion Engineering, Inc. (C-E) and KWU have indicated that substantial reductions in the extent of in-reactor densification can be achieved by increasing the initial density of the fuel, use of fabrication methods leading to stable microstructures, or by a combination of the two approaches. Stable fuel has been produced and irradiated with little or no change in density. In addition, pre-pressurized fuel rod designs have been adopted in which the initial fill pressure is adequate to avoid cladding collapse during the full in-reactor lifetime of the fuel.

This paper summarizes the results of in-reactor tests designed to study the extent and kinetics of densification for fuels with various microstructures and initial densities. In addition, a review of mechanisms associated with in-reactor densification is included to provide a link between theory and experimental observations showing the dependence of densification on UO_2 microstructure.

Thermal sintering of ceramic compacts that are near theoretical density occurs by the elimination of pores. Pore removal occurs by the gradients in vacancy concentration that are established between the curved pore surface and the flatter grain boundary surfaces. A vacancy flux between these regions is equivalent to a net transfer of atoms from the grain boundary to the pore, which results in a net increase in density.

Out-of-pile densification kinetics have been treated with both bulk and grain-boundary diffusion as the rate controlling processes.[2,3] Both of these diffusion processes are thermally activated with the result that sintering rates increase markedly with increasing temperature. Uranium dioxide requires temperatures in excess of 1200°C before out-of-pile sintering kinetics are rapid enough for microstructural changes to be observable in a reasonable time.

(Cont'd pg 85)

In-reactor densification of UO_2, however, has been observed in relatively short times, at temperatures (400 to 1000°C) which are much lower than those required out-of-pile.

The removal of small pores in UO_2 fuel as a result of low temperature irradiations for relatively short periods of time has been observed by a number of investigators.[4-8] Pores affected by this phenomenon include both fission gas bubbles and the closed porosity, initially present in the as-sintered material.

The occurrence of densification and pore annihilation at low temperatures requires significant irradiation enhancement of the rate of material transport associated with pore annihilation. The two most probable mechanisms, which have been suggested as the cause of pore annihilation and irradiation-induced densification, are:

1. *Fission spike overlap:* Fission fragments that pass close to the surfaces of a pore aid in trapping vacancies in a deposited layer of fuel when material is blasted or sputtered from one side of the pore to the other. Some of these vacancies can migrate away from the pore, resulting in mass transfer of atoms to the pore and eventual disappearance of the pore. This mechanism has been notably described by Stehle and Assmann.[9,10]

2. *Enhanced diffusion-related mechanisms:* The increased vacancy concentration produced by the fission event results in increased uranium diffusivities. The higher diffusion rates enhance classical sintering mechanisms.

An experimental fuel irradiation program on densification has been performed in the Battelle Research Reactor (BRR) at linear heat ratings from 90 to 150 W/cm. The fuel types included microstructures predicted to be both stable and unstable. The unstable microstructures were fine grained (<6 μm) and contained a large fraction of their pore volume in pores <4 μm in diameter. The stable fuel microstructures included a combination of large grain size and/or large pore size. The grain sizes ranged from 10 to 15 μm, and the pores sizes ranged from 1 to 100 μm. The stable fuels had small fractions of their pore volumes in sizes below 4 μm. The kinetics of densification were measured up to 1500 h. As shown in Fig. 1, the unstable fuels densified rapidly, while the stable fuels showed little change in density. Results from the Halden tests shown below indicate that almost all the densification occurs prior to this exposure. Among the unstable fuels, a general trend toward increased densification with lower initial density was noted.

In a joint C-E/KWU test at Halden, instrumented fuel rods are utilized to determine the dynamic response of fuels with various stabilities during operation at 300 and 500 W/cm. The initial densities range from 92 to 96% TD and exposure for the higher powers was 15,000 MWd/MTU. Two of the C-E fuel rods contain elongation sensors which measure stack length changes relative to the cladding. These in-reactor, fuel-column length data provide information of the kinetics of densification as a function of burnup and heat rating. Consistent with the BRR test results, these fuels densified rapidly early in the irradiation and the rate of densification dropped

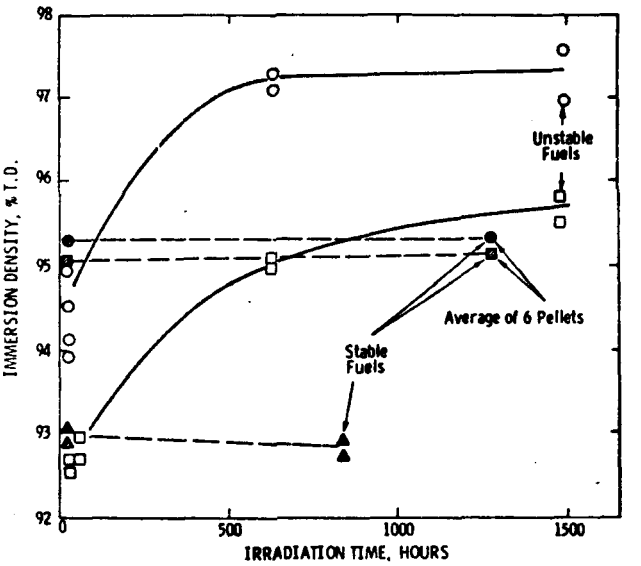


Fig. 1. UO_2 pellet density vs irradiation time for stable and unstable fuel (90 to 150 W/cm).

sharply as the irradiation progressed. Two fuel types used in the rods with elongation sensors included microstructures predicted to be stable. Densification kinetics are shown in Fig. 2. After irradiation to burnups of 5000 MWd/MTU, the two C-E fuel rods show column length shrinkages of only 0.22 and 0.18%. Assuming isotropic densification, volumetric shrinkages of the fuels are only 0.66 and 0.54%, respectively. Thus, the terminal densification in the stable fuel types, defined as the densification which occurs in the first 4000 MWd/MTU burnup, averages ~0.6%. As shown in Fig. 2, the terminal density value is essentially attained within the first 1000 MWd/MTU. The expected densification for the newer C-E fuels is, therefore, 0.6% or less.

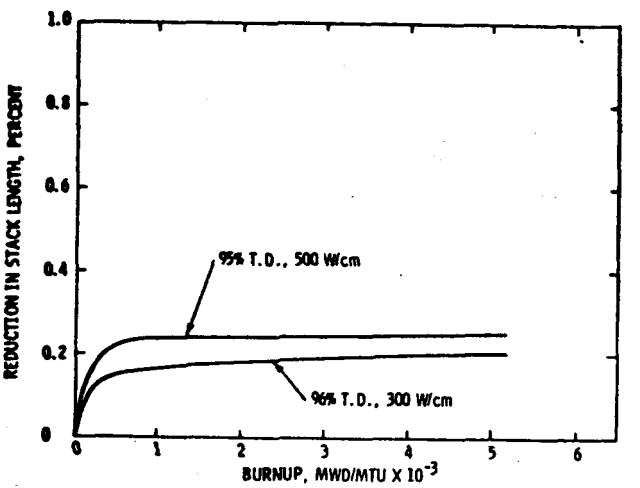


Fig. 2. Densification kinetics of C-E stable fuels irradiated in Halden.

(Cont'd pg 86)

1. "Technical Report on Densification of Light Water Reactor Fuel," Regulatory Staff, USAEC (Nov. 1972).
2. R. L. COBLE, "Sintering Crystalline Solids, 1. Intermediate and Final State Diffusion Models," *J. Appl. Phys.*, 32, 5, 787 (1961).
3. R. L. COBLE and T. K. GUPTA, "Intermediate Stage Sintering," in *Sintering and Related Phenomena*, G. C. LUCZYNSKI, H. A. HOOTEN, and C. F. GOBBON, Eds., Gordon and Breach, New York (1967).
4. A. D. WHAPHAM, "Electron Microscope Observation of the Fission Gas Bubble Distribution in UO_2," *Nucl. Appl. Technol.*, 2, 123 (1966).
5. A. M. ROSS, "Irradiation Behavior of Fission Gas Bubbles and Sintering Pores in UO_2," *J. Nucl. Mater.*, 30, 134 (1969).
6. J. A. TURNBULL and R. M. CORNELL, "The Re-Solution of Gas Atoms from Bubble During the Irradiation of UO_2," *J. Nucl. Mater.*, 36, 161 (1969).
7. J. A. TURNBULL and R. M. CORNELL, "Observations Demonstrating the Re-Solution of Gas from Bubbles and Sintering Pores During the Irradiation of UO_2 at High Temperature," *J. Nucl. Mater.*, 37, 355 (1970).
8. J. A. TURNBULL and R. M. CORNELL, "The Re-Solution of Fission-Gas Atoms from Bubbles During the Irradiation of UO_2 at an Elevated Temperature," *J. Nucl. Mater.*, 41, 156 (1971).
9. H. ASSMANN and H. STEHLE, "Verdichtungseffekte in Gesintertem UO_2," Reaktoragung, Karlsruhe, *Tagungsoright*, p. 409 (1973).
10. H. STEHLE and H. ASSMANN, "The Dependence of In-Reactor UO_2 Densification on Temperature and Microstructure," *J. Nucl. Mater.*, 52, 303 (1974).

ANS TRANS., vol. 20, 1975

2. The Influence of Fuel Microstructure on In-Pile Densification, *W. Chubb, A. C. Hott, B. M. Argall, G. R. Kilp (W-NES)*

Recognition of the phenomenon of densification was followed by a review of extensive Westinghouse post-irradiation data. In general, the pre-irradiation fuel structure was known from characterization or archive samples. Particular emphasis was placed on the grain size and pore morphology (amount, size, and spatial distribution), since analyses had determined that these would be controlling factors for the rate and extent of densification.

Among the Westinghouse results available at this time was a parametric study of the effects of fuel density, fuel-cladding gap, and rod internal pressure on fuel rod performance. Fuel for these rods, for the most part, had been made by the standard Westinghouse fabrication process, but the varying densities were achieved by sintering over a several hundred degree temperature range. These fabrication differences gave rise to structure differences (primarily grain size and extent of fine porosity) as well as the required difference in density. Several rods containing low- and intermediate-density fuel were also made by an alternate pellet fabrication route; this fuel exhibited a markedly different structure compared to the low-density fuel fabricated by the standard process. As a result, porosity distribution effects could be evaluated. The effect of the modified process in terms of structure and resultant densification for the low-density fuels is summarized in Table I. Most of the benefit is attributed to the significant reduction in fine porosity achieved through process modification, although the grain size is somewhat larger for the modified process fuel.

The pre-irradiation grain size and extent of fine porosity generally correlated well with sintering temperature despite the process differences noted. The fuel with larger grain size and coarser porosity was shown by postirradiation evaluation to result in improved stability. Because reduced total porosity, reduced fine porosity, and increased grain size were general trends with increasing sintering temperature in these rods, the separate effects of each were not easily distinguishable. As an example, our investigations indicated that pores in a high-density fuel sintered at 1650°C are more stable than the same size pores in a lower density fuel sintered at 1450°C. In this case, it was not possible to separate the effect of grain size, density, or sintering temperature as the primary variable influencing pore stability. Such separation depended on a broader data base than provided by available postirradiation data.

TABLE I

Dependence of Densification on Fuel Structure
89.5% TD UO_2

Fuel	Sintering Temp (°C)	Porosity (<5μm) (%)	Grain Size (μm)	Density Change (%)
Standard Process	1450	6.8	2.5	7.5
Special Process	1750	3.2	5.7	1.8

Because of the limited number of evaluations possible and the time delays inherent in destructive evaluation, a modification was made to the application of existing in-core instrumentation to permit fuel stack length determinations to be made on a periodic basis. This measurement technique was then applied to many operating fuel regions. Through review of archival data and fabrication records the effects of the various structure parameters were evaluated. Because there was more independence of density, grain size, and sintering temperature for these data, it was possible to get a separation of variables better than that achieved by destructive postirradiation evaluation. At the same time, analysis of over 500 individual measurements of densification revealed that such separation of variables was largely of academic interest. Increased grain size and reduced fine porosity levels, both of which favor fuel stability, were found to vary consistently with higher sintering temperatures for the Westinghouse standard process. Thus, for a given final density, control of the sintering parameters of time and temperature provided a good measure of control on grain size and fine porosity level.

(Cont'd pg 87)

Evaluations of all data developed to date indicate that the amount of fine porosity present in the fuel is dominant in determining the extent of densification. For a given level of fine porosity, some improvement is afforded by increased grain size. Both of these are effectively controlled in the standard Westinghouse process where density and sintering conditions—time and temperature—are defined. Detailed evaluations of our data suggest a significant incremental benefit of high sintering temperatures, *per se*, over and above the benefits attributable to apparent improvements in fuel structure.

ANS TRANS., vol. 18, 1974

JOURNAL OF NUCLEAR MATERIALS 21 (1967) 77–87. © NORTH-HOLLAND PUBLISHING CO., AMSTERDAM

DIRECT OBSERVATION OF FISSION FRAGMENT DAMAGE IN SOME CERAMIC OXIDES

J. H. CHUTE *

Australian Atomic Energy Commission, Research Establishment, Lucas Heights, Australia

Received 29 July 1966

Fission fragment damage in thin flakes and films of BeO, ThO_2, and Al_2O_3 has been studied by transmission electron microscopy. No fission tracks were found in Al_2O_3 and only occasional tracks were detected in BeO; however, well-defined tracks were present in ThO_2. The nature of the tracks is consistent with an ionization process as the main mechanism of energy loss. Although nuclear collisions do occur, such events are comparatively rare.

17359 AEC STUDY ON THE DENSIFICATION OF NUCLEAR FUEL. Com. Naz. Energ. Nucl., Notiz.; 19: No. 2, 80-82(Feb 1973). (In Italian).

An account of a study conducted by the AEC on the densificati[on] of the nuclear fuel used in LWR's is presented. This study was started in May 1972 after the flattening of small sections of the unpressurized fuel rods in the Ginna Plant reactor. (auth)

NSA, vol. 28, 1973

17547 DENSIFICATION OF REACTOR FUELS. Culbert, W. H. Nucl. Safety; 14: No. 4, 356-361(1973).

The movement of fuel as a consequence of densification was determined to be the cause for "collapse" of the fuel cladding in a number of pressurized-water reactors. Cladding collapse presented at unevaluated configuration for heat removal in both normal and accident situations. On Nov. 14, 1972, the Regulatory Staff of the U. S. Atomic Energy Commission issued its Technical Report on Densification of Light-Water Reactor Fuels. On Feb. 2, 1973, the Staff issued the report entitled Additional Testimony on Point Beach 2 Nuclear Plant in Regard to Fuel Densification and Its Effects. Summaries of both of these AEC reports are presented along with the bibliography included with the November 14 report. (auth)

NSA, vol. 28, 1973

UO_2 Densification—Irradiation Results, Laws, and Models, *Ghislain de Contenson, Guy Lestiboudois, Nicole Vignesoult (CEA—France)*

INTRODUCTION

Irradiation carried out by the CEA in experimental and power reactors revealed oxide column retractions already in 1970, resulting from densification of the UO_2.

However, this phenomenon has never had secondary consequences shortening the life of the fuel elements.

Some pins do occasionally have inter-pellet play one-third up the stack without reduction of the active column. Such phenomena do not appear to be ascribable to densification, but to a UO_2-cladding adherence through the fission products, since they only appear as from 10,000 MWd/MTU.

Nevertheless, specific studies of the densification phenomenon were undertaken, as a result of which:

1. The radial and longitudinal densification kinetics of different types of fuel were elucidated.
2. An out-of-pile test representative of in-pile densification was perfected.
3. A densification model was suggested to predict the behavior of fuel elements.

STUDY OF DENSIFICATION KINETICS

A distinction has to be made between longitudinal and radial densification.

Longitudinal Densification

Two methods were used simultaneously:

1. Continuous measurement of the internal pressure: The pressure drop observed at the beginning of irradiation corresponds to the voluminal increase in the expansion chamber through the reduction in the height of the stack (TOF 38 to 44 experiments).

2. Metrology of the length of the oxide column by successive neutron radiographies during irradiation (Anemone experiments).

Radial Densification

This measurement is not effected directly but through a temperature determination in the center of the fuel by a W-WRe thermocouple (Irène experiments). From the linear power and knowing the core temperature of the oxide and the outside temperature of the cladding, the transfer coefficient may be calculated. A variation in the central temperature at constant power therefore corresponds to a variation in the transfer coefficient.

A law (transfer coefficient—play when hot) established by the CEA (Furet experiments) enables the temperature variations to be expressed as play-when-hot variations; therefore, the radial densification.

Results

These TOF, Anemone, and Irène experiments enabled us to determine the variations with time of the ratio, d/a, where

d = variation in the volume at moment, t

a = variation of the possible maximum volume deduced from the out-of-pile test.

Figures 1 and 2 show the time-power diagram for determining the ratios, d/a, longitudinally and radially.

CHOICE OF STABILITY TEST AND REPRESENTATIVITY (20 h at 1720°C)

The 20 h at 1720°C annealing test in hydrogen was chosen further to a certain number of comments relating either to the sintering or to the in-pile performance of the UO_2 pellets.

1. The industrial sintering of stable-tendency UO_2 is generally treated at a temperature of around 1650°C maintained for several hours—say, 2 to 5. The plateau temperature plays a more important role in the densification than the holding time at this temperature.

In accord with other teams, it seemed logical to choose a higher annealing temperature but still industrially acceptable without too much difficulty; i.e., around 1720°C or 2000 K.

The annealing time at this temperature must not be too long for practical working reasons, and 20 h was deduced from a densification kinetics study of several kinds of samples, previously sintered at 1650°C at least.

2. The simultaneous in-pile experiment showed that for the different categories of sintered UO_2, this stability test at 1720°C could be considered as representative of the packing and the diametral reduction of the fuel observed under neutron flux because:

(a) At first, the reductions of "fuel" height, $\Delta h/h$, of many of the pins irradiated in power reactors are in the 0.6 to 1.2% bracket and the out-of-pile densification test (20 h at 1720°C) of the same UO_2 batches gives a 3% average reduction in volume or a 1% $\Delta h/h$.

(b) Second, a specific test of the reduction in length (Anemone program) of pins made with UO_2 pellets of conventional porosity and with stable porosity pellets established a parallel between in-pile and out-of-pile kinetics. This 20-h test at 1720°C gives the "A" value of maximum possible volume variation for a fuel.

This is used as basic data in the densification model

$$a = 1.2 \left[\left(\frac{\Delta V}{V} \right)_0^{20\ h} \right]_{1720°C} .$$

DENSIFICATION MODEL

As from the results obtained, a physical outline of the densification phenomenon was made in order to introduce it into a thermal and mechanical evolution calculation code. This "densification under irradiation" adds to the "thermal densification" already allowed for in the code.

Basic Model

The separation between the two types of radial densification is made both sides of the 1750°C isotherm. It is considered that the pores in that part of the fuel external to the isotherm migrate to the exterior, while those in the

(Cont'd pg 89)

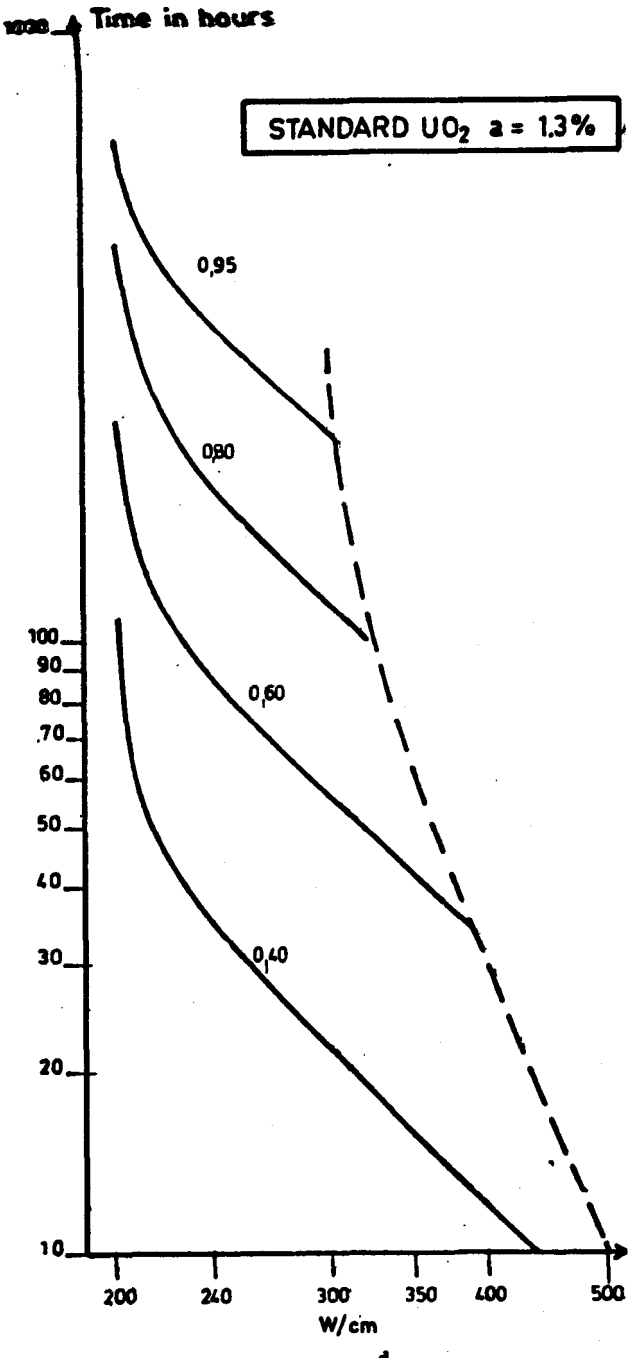


Fig. 2. Diametral densification $\frac{d}{a}$ with time and power.

internal part of this isotherm continue to migrate to the center with formation of a central void. The radial densification under irradiation increases the play between the fuel and the cladding.

Densification under irradiation also reduces the length of the fuel column. It is considered that the pores migrate to the ends of the column.

Formulation of the Correlation

The correlation established expresses radially and axially the relative densification obtained according to the irradiation parameters: specific power and irradiation time.

Relative densification means the ratio between the densification sought (d) and the total possible voluminal densification rate defined out-of-pile by a 20-h test at 1720°C (a):

$$a = 1.2\left[\left(\frac{\Delta V}{V}\right)_0^{20\text{ hours}}\right] \text{ at } T = 1720°C \ .$$

To attempt a generalization of the formulation, the number, F, of fissions per cm^3 was introduced:

$$F = \frac{f}{\pi \times R^2_{UO_2}} \times Pl \times t \ ,$$

where

Pl = linear power in W/cm

t = time elapsed since the start of irradiation (in hours)

$f = 3.14 \times 10^{10}$ fiss/J,

and a grouping of parameters of form $X = F/t^{\alpha}$ was chosen.

The correlation obtained, valid radially and axially, is therefore

$$\frac{d}{a} = \text{th}\left[k(\beta x + \gamma x^2 + \delta x^3)\right] \ .$$

The constants calculated from the experimental results are:

	α	β	γ	δ	k
Radial densification	0.55	1.97×10^{-18}	-4.28×10^{-37}	4.57×10^{-56}	2
Axial densification	0.44	0.468×10^{-18}	1.57×10^{-37}	1.23×10^{-56}	0.5

Results

Comparison of the estimated figures and the results of the experiments show that the formulation adopted qualitatively reflects the physical phenomena observed.

CONCLUSIONS

These studies have shown that not only is there longitudinal densification but also radial densification which could cause a temperature increase of 200°C in the center of the UO_2 for linear power of 400 W/cm. Densification laws have been sketched and a model established. This model has been added to the calculation code of fuel element behavior. It fully accounts for the phenomena observed.

Laws of pore elimination by densification were also established after irradiation on irradiated fuel. These laws showed that virtually all the pores in the 0.3- and 2-μm bracket disappear through densification between 800 and 1500°C.

1543 (NEDM-10735) DENSIFICATION CONSIDERATIONS IN BOILING WATER REACTOR FUEL DESIGN AND PERFORMANCE. Ditmore, D. C.; Elkins, R. B. (General Electric Co., San Jose, Calif. Nuclear Fuels Dept.). Dec 1972. 48p. Dep. NTIS $4.50. The fuel densification mechanism and the potential effects of this mechanism on the performance of GE BWR Zircaloy clad UO_2 pellet fuel are analyzed. 14 references. (auth)

NSA, vol. 28, 1973

7. Fuel Performance Experience in C-E and KWU Pressurized Water Reactors, *R. N. Duncan, W. P. Chernock (CE), D. Knodler, H. Stehle, H. Knaab (KWU)*

Fuel performance experience to date in C-E and KWU pressurized water reactors has been very satisfactory. As a result of recent C-E/KWU agreements on PWR technical information exchange, detailed fuel performance data can be reported for the three operating PWRs listed in Table I.

The Palisades and Maine-Yankee reactors utilize Zircaloy-4-clad UO_2 fuel with pellet densities of ~93 to 94% TD. All the fuel assembly structural components within the core region are Zircaloy-4, including the grids.

Fuel performance in both reactors has been excellent thus far at maximum authorized interim power levels (Palisades 100%, Maine-Yankee 75%). The fission product activity in the primary coolant, shown in Table II, has been low since the startup of these reactors. The fuel has achieved burnups of over 6000 MWd/TU in the Palisades reactor and over 2000 MWd/TU in the Maine-Yankee reactor.

These extremely low iodine values substantiate the contention that attention to manufacturing detail and quality control, for conservative fuel designs, results in a very low incidence of fuel rod defects in entire core loadings.

The KWO reactor is currently operating in its 4'th cycle. The core contains Zircaloy-4-clad UO_2 fuel fabricated with pellet densities of 94 to 95% TD. Inconel is used for the grids and support structure.

The first KWO core, which contained unpressurized fuel rods, has operated three full cycles to burnups over 35,000 MWd/TU average without evidence of life limit failures.[1] One first-core assembly is still in the reactor for a 4'th cycle as a high burnup experiment. All KWO reload fuel is pressurized and has successfully achieved two full cycles of operation to exposures of about 21,000 MWd/TU.

During the operation of KWO cycles 1 and 2, fission product activity in the primary coolant increased.[1] Overall, there has been a high degree of flexibility in the selection of alternate reload patterns for KWO to achieve economic and safe fuel management.

Extensive fuel examinations of both the initial core and reload assemblies indicate that no fuel rod cladding collapses occurred in any of the KWO fuel and there were no other observed anomalous manifestations such as gaps in the fuel column due to possible UO_2 fuel densification.[1] The potential for in-reactor densification of UO_2 was recognized early by KWU and resulted in successful optimization of both UO_2 density and microstructure.[2] Gamma scans of fuel rods after one cycle of operation indicated axial fuel stack length decreases of ~0.1% with a maximum of 0.3%.

TABLE I

C-E and KWU Pressurized Reactors

Reactor	Supplier	Rating MW(e)	Generating Power Since
KWO (Kraftwerk Obrigheim)	KWU (Kraftwerk Union PWR Division)	340	3/69
Palisades	C-E	720	8/72
Maine-Yankee	C-E	830	12/72

TABLE II

Primary Coolant Iodine Activity

Reactor	Percent of Rated Power	Steady-State Coolant Activity ($\mu Ci/cm^3$)	
		^{131}I	^{133}I
Palisades	80%	2.1×10^{-3}	4×10^{-3}
Maine-Yankee	75%	2.5×10^{-4}	5×10^{-4}

1. D. KNODLER, H. SCHENK, and H. STEHLE, "Irradiation Performance of Zry-Clad Fuel Elements of Obrigheim PWR Plant," presented at Technical Meeting 5/1, NUCLEX (October 21, 1972).
2. H. ASSMAN, H. KROLL, and H. ROEPENACK, "Preparation and Properties of Lower Density UO_2 Pellets," AED CONF-71-100-27 (May 1971), *Proc. U.N. Intern. Conf. Peaceful Uses At. Energy, Fourth, Geneva* (September 1971).

ANS TRANS., vol. 16, 1973

15617 (BAW-10084) PROGRAM TO DETERMINE IN-REACTOR PERFORMANCE OF B AND W FUELS. Cladding Creep Collapse. Eckert, A. F. J.; Wilson, H. W.; Yoon, K. E. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Nuclear Power Generation Div.). Nov 1974. 58p.

The determination of the creep rate for B & W Zircaloy-4 fuel rod cladding is described. The development of the computer code, the verification with experimental data, and the development of the collapse criteria are presented. The use of the analytical method in analyzing power reactors is discussed. A sensitivity study using the CROV code is given. The conservatisms in the analytical procedure are discussed. (auth)

NSA, vol. 31, 1975

27951 THE BEHAVIOR OF BWR FUEL ELEMENTS DURING POWER OPERATION. Principles and Data. Eickelpasch, N.; Seepolt, R. W. (Kernkraftwerk RWE-Bayernwerk G.m.b.H., Gundremmingen/Guenzburg (F.R. Germany)); Wolff, U. (Rheinisch-Westfaelisches Elektrizitaetswerk A. G., Essen (F.R. Germany). Abt. Kraftwerke). At. Strom; 20: No. 9/10, 99-108(Oct 1974). (In German).

22 figs.; 27 refs.

Methods are described for inspecting fuel elements both during reactor operation and outside the reactor. By using examples of fuel elements in the nuclear station at Gundremmingen the evaluation and the results from the more important methods are presented. (auth)

NSA, vol. 31, 1975

8220 (EPRI-NP-2) NUCLEAR POWER DIVISION: RESEARCH PROGRESS REPORT NP-2. (Electric Power Research Inst., Palo Alto, Calif. (USA)). Feb 1975. 31p. NTIS $4.00.

Research progress in areas concerning reactor safety, fuel densification, plutonium recycle, fuel element testing, stress corrosion, etc., is reported. (LTN)

NSA, vol. 32, 1975

10739 (XN-209) DENSIFICATION EFFECTS ON EXXON NUCLEAR PRESSURIZED WATER REACTOR FUEL. (Exxon Nuclear Co., Richland, Wash. (USA)). Mar 1974. vp. Exxon Nuclear Co., Inc., Richland, WA.

Individual reports are included on the following subjects: power spike model for PWR fuel; cladding collapse calculational procedure; GASPRX calculation procedure for internal gas pressure due to fission gas release; pellet-to-cladding gap closure from pellet cracking; and GAPEXX program for predicting pellet-to-cladding heat transfer coefficients. (JWR)

NSA, vol. 32, 1975

13147 (XN-209(Suppl. 4)) DENSIFICATION EFFECTS ON EXXON NUCLEAR PRESSURIZED WATER REACTOR FUEL. Supplement 4. (Exxon Nuclear Co., Richland, Wash. (USA)). 22 Nov 1974. 21p.

A description is given of modifications to the calculational procedures incorporated in the PWR densification model to account for a threshold in fuel pellet burnup below which irradiation-induced UO_2 swelling does not affect the pellet dimensions, and the potential for absorption of helium fill gas into the UO_2 fuel pellets. Analytical results are also provided which compare fuel and clad temperatures predicted using the computer codes GAPEX and JEREPP. The input data used in the calculations and a general description of the JEREPP code calculational technique are also provided. (auth)

NSA, vol. 31, 1975

18750 (XN-209(Suppl.4)(Rev.1)) DENSIFICATION EFFECTS ON EXXON NUCLEAR PRESSURIZED WATER REACTOR FUEL. Supplement 4 (Revision 1). (Exxon Nuclear Co., Richland, Wash. (USA)). 27 Dec 1974. 28p. Exxon, Richland, WA.

Modifications to the calculational procedures incorporated in the PWR densification model are described which account for: (1) a threshold in fuel pellet burnup below which irradiation-induced UO_2 swelling does not affect the pellet dimensions, and (2) the potential for absorption of helium fill gas into the UO_2 fuel pellets. Analytical results are also provided which compare fuel and clad temperatures predicted using the computer codes GAPEX and JEREPP. The input data used in the calculations and a general description of the JEREPP code calculational technique are also provided. A comparison of equiaxial grain growth and percent fission gas release calculated using GAPEX to three different experimental values is provided. In each case the comparison indicates that calculations using GAPEX conservatively overpredict the fuel temperature. (auth)

NSA, vol. 31, 1975

1. Nuclear Fuel Experience in Westinghouse Pressurized Water Reactors, *Harry M. Ferrari (W-NES)*

As of February 1973, nuclear fuel designed and fabricated by Westinghouse had generated over 110 billion kWh of electric energy. This experience was gained with a variety of fuel designs in 18 operating nuclear power plants.

During the early years of commercial nuclear power, stainless-steel fuel rod cladding was used in most Westinghouse fuel. The experience in all cases—Yankee, Indian Point 1, Haddam Neck, San Onofre, Chooz, and Trino—was excellent. Of the ~200,000 stainless-steel-clad rods irradiated, the overall defect level is estimated to be only about 1/10,000 rods.

The advent of private ownership of nuclear fuel resulted in fuel cycle economics which favored Zircaloy. Starting with Zorita in 1968, all Westinghouse-designed fuel for new cores used Zircaloy. As of January 1973, over 300,000 Westinghouse-designed and -fabricated Zircaloy-clad fuel rods were operating in 12 large power reactors.

The experience with Zircaloy-clad fuel in large pressurized water reactors has been very good, although several abnormalities were observed in early unpressurized Zircaloy fuel designs.

Relatively high coolant activity developed in Beznau I and Ginna during the first few months of operation. The problem was limited to the outer, lowest density region of Beznau and Ginna fuel. Evaluation of the fuel leakers led to the conclusion that it was caused by excess moisture in the fuel, which then caused numerous local cladding hydriding defects. Design and fuel rod fabrication procedures were changed in 1969 to reduce and control moisture levels in the fuel. Operating experience in seven reactors that have started up with fuel fabricated since these changes has shown very low activity at times well beyond the point where the Beznau and Ginna plants developed high activity, thus demonstrating the localized hydriding is no longer a problem.

During Cycle 1 refuelings of Jose de Cabrera and Ginna reactors, a large number of fuel rods were observed to be in interference with the top nozzles, and a few of these rods were bowed. The rod interference and bowing was due to larger-than-expected Zircaloy growth during irradiation. Although some axial growth had been considered in design, it was based on limited data on Zircaloy material having a different texture (preferred grain orientation) and proved to be inadequate in the earliest designs. This problem was easily remedied by adopting a more conservative Zircaloy growth model and increasing the axial gap between fuel rods and nozzles.

The most recent abnormality was the observation of a number of axial fuel pellet gaps and locally flattened sections in early unpressurized Zircaloy fuel designs in Beznau I and Ginna. The phenomenon was due to densification of the UO_2 fuel during irradiation and axial shrinkage of the UO_2 pellets. The axial pellet shrinkage combined with random pellet hangup due to fuel-cladding interaction resulted in gaps occurring in the fuel column when other pellets settled in the fuel rod. These gaps varied in length and location. In some cases where the axial gap was large, cladding creep occurred due to the high external coolant pressure into an oval configuration and eventually flattened due to elastic instability.

Cladding flattening as occurred in early unpressurized fuel designs in Beznau I and Ginna can be avoided by pre-pressurizing the fuel prior to irradiation[1] to sufficiently high levels. All Westinghouse Zircaloy fuel delivered since 1969 has been pressurized, and all Westinghouse Zircaloy fuel remaining in operation after mid-1973 will be pre-pressurized.

Although considerable adverse publicity occurred because of fuel densification and cladding flattening, the number of flattened fuel rods was small; of those flattened, the number defected was smaller still, and in no cases did the coolant activity levels approach operating limits.

Although fuel densification under irradiation has been a problem, extensive studies performed during the past 18 months have resulted in an understanding of the problem. Data from several operating reactors show that Westinghouse fuel fabricated within the past 2 to 3 years is relatively stable (densification <1 to 2%). Recent changes in specification will result in even more stable fuel. As a result, densification is no longer considered a problem.

Although several abnormalities have occurred with Zircaloy-clad UO_2 fuel in pressurized water reactors, the difficulties were all soluble. The extremely low coolant activity in the 12 operating reactors which have Zircaloy (estimated at 2/10,000 rods as of February 1973) attests to the reliability of current Zircaloy-clad fuel in Westinghouse pressurized water reactors.

1. H. M. FERRARI, "Internally Pressurized Zircaloy-Clad Fuel Rods," *Trans. Am. Nucl. Soc.*, 12, 554 (1969).

ANS TRANS., vol. 16, 1973

13096 FUEL DENSIFICATION EXPERIENCE IN WESTINGHOUSE PRESSURIZED WATER REACTORS. Ferrari, H. M.; Roberts, E.; Scott, J. (Westinghouse Electric Corp., Pittsburgh). pp 54.1-54.4 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004-.

Although fuel densification under irradiation has probably occurred in most oxide fuel irradiated to date, the significance of this phenomenon became apparent only after the fuel densification and collapse observations in Beznau I and Ginna. Based on intensive evaluation of irradiated fuel fabricated by different processes to different specifications, many aspects of the fuel densification phenomenon are now becoming clear. Densification results primarily from the disappearance of fine porosity. It occurs rapidly and is essentially complete within a few thousand hours. Fuel densification is highly dependent on initial fuel fabrication processes and pellet microstructure. The phenomenon can be minimized by proper control of pellet microstructure. Clad flattening as occurred in early unpressurized fuel designs such as Beznau and Ginna can be avoided by increasing the initial helium pre-pressurization levels. All operating Westinghouse fuel after mid-1973 will be pre-pressurized. (14 references) (auth)

NSA, vol. 32, 1975

3. Thermal Reactor Mixed-Oxide Fuel Performance, *M. D. Freshley (PNL)*

The universal acceptance of the demonstrated basic technical feasibility of plutonium utilization in thermal reactors attests to the success of numerous experimental programs that were devoted to the achievement of this goal during the past decade (Table I).[1]

Irradiations in PRTR were characterized by testing a variety of different mixed-oxide fuel types, some of which may be considered to be advanced concepts, to significant burnups (over 18,000 MWd/MTM) at linear heat ratings (about 20 kW/ft peak) higher than those currently employed in commercial power reactors. The different mixed-oxide fuels tested included Zircaloy-clad, vibrationally compacted and swage compacted, heterogeneously and homogeneously enriched types, as well as pellet fuels produced by hot pressing and the more conventional cold-press-sinter technique.[2] All fuel types performed satisfactorily.

Although small PuO_2 concentrations in mixed-oxide thermal reactor fuel have an insignificant effect on most irradiation performance characteristics, one of the objectives of the mixed-oxide fuel development program at Pacific Northwest Laboratories was to investigate important considerations, unique with plutonium-enriched fuels, that are related to performance.

"Free-standing" Zircaloy-2 cladding becomes oval shaped because of creep collapse during the early stages of irradiation in PRTR.[3] Ovality is more pronounced in mixed-oxide rods containing cold-press-sinter pellet fuel than in rods containing hot-press pellets and Vipac fuel.

Sorbed gases and moisture released from mixed-oxide fuel during irradiation react rapidly with the Zircaloy cladding and do not contribute to the internal pressure within the rod.[4] Fission gas release in Vipac rods varies linearly with volumetric average fuel rod temperature. For comparable volumetric fuel rod temperatures, the fission gas release fraction from cold-press-sinter fuel is less than from Vipac fuel, whereas the release fraction from hot-press pellet fuel is greater than from Vipac fuel.

Mechanically mixed UO_2-PuO_2 fuel including pneumatically impacted Vipac, hot-press pellet, and cold-press-sinter pellet, all contain localized or micro PuO_2 nonuniformities. Homogenization commences at temperatures sufficient to cause sintering and equiaxed grain growth and uniform solid-solution formation occurs rapidly above columnar grain-growth temperatures. Evidence of significant fission product migration in mixed-oxide fuels commences at maximum fuel temperatures above 1800°C and at maximum fuel temperatures above 1200 to 1300°C, cesium burnup determinations are inaccurate because of migration.

A significant change in radial plutonium concentration can occur rapidly in mixed-oxide fuels during irradiation.[5] Radial plutonium redistribution is coincident with the columnar grain-growth restructuring phenomenon and occurs as a result of the preferential evaporation of UO_2-rich vapor from the central hot region of the fuel to the cooler peripheral regions near the cladding.

A reaction occurs between the ZrO_2 layer on the inner cladding surface in hyperstoichiometric hot-press pellet and Vipac mixed-oxide fuel. In those instances where fuel/cladding reaction occurs, the ZrO_2 layer becomes porous and separates, which could have a significant effect on fuel/cladding heat transfer.

Defect behavior is an important mixed-oxide fuel performance criterion. Although the defect performance of mixed-oxide fuels has been satisfactory, there is some evidence that suggests the defect mode may change at high linear heat ratings and could become the limiting consideration.[6]

The possible effect of PuO_2 particle size and distribution on the transient behavior of mixed-oxide fuels is also an important performance consideration. Single 550-μm-diam PuO_2 particles dispersed in the oxide fuel reduces the threshold energy for cladding perforation. However, no indications of the effects of prompt fuel dispersal due to the expulsion of the PuO_2 particles into the surrounding water are evident.[7]

The results of these experimental programs have culminated in the initiation of demonstration programs in commercial power reactors (Table I), with the rapid full-scale implementation of plutonium utilization in commercial plants anticipated in a timely manner. Because of the current state of the industry, cold-press-sinter pellet fuel is presently the reference mixed-oxide fuel type for commercial application. However, vibrationally compacted fuel is considered to be the most promising alternate fuel form.

Although there do not appear to be any performance limitations unique to mixed-oxide fuels, for ease of licensing the first mixed-oxide fuel designs introduced into commercial plants on a significant scale will undoubtedly be based on proven technology and plutonium enrichment will be simply substituted for ^{235}U. One of the primary concerns relates to fuel fabrication quality assurance as related to PuO_2 particule size and PuO_2 particule distribution requirements in the absence of experimentally determined size and nonuniformity limits.

To better optimize plutonium utilization, future fuel designs that exploit the unique characteristics of plutonium to improve power sharing among rods in fuel assemblies, extend reactivity lifetime, and further improve fuel operating characteristics must be developed.

TABLE I

Significant Domestic Mixed-Oxide Irradiations

Reactor	No. of Rods	Fuel Type[a]	Peak Burnup (MWd/MTM)[b]	Peak Rod Power (kW/ft)	Peak Heat Flux [Btu/(h ft^2)]
PRTR	~1860	Swage	13,500	17	390,000
PRTR	~700	Vipac	18,500	16	370,000
PRTR (High Power Density)	~1730	Pellet (cold-press-sinter and hot-press)-Vipac-Swage	13,000	21.5	500,000
EBWR	1296	Vipac	3,000	8	200,000
Saxton-Core II	638	Pellet-Vipac	29,000	16	530,000
Saxton-Core III	~250	Pellet	44,000	19	630,000
Dresden-I[c]	4	Pellet (hot-press)	13,000	10	230,000
Dresden-I[c]	99	Pellet	14,800	14.9	345,000
San Onofre[c]	720	Pellet	8,500	13	460,000
Big Rock Point[c]	32	Pellet	19,500	14	325,000
Big Rock Point[c]	204	Pellet	14,600	16	370,000

[a]Pellets are cold-press-sinter type unless otherwise noted.

[b]Megawatt-days per metric tonne of metal projected to September 1971.

[c]Commercial power reactors.

1. F. G. DAWSON, "Thermal Reactor Plutonium Fuel Irradiation Programs in the United States," BNWL-SA-3855, IAEA Panel on Plutonium Recycling in Thermal Power Reactors, Vienna, Austria (June 21-25, 1971).

2. M. D. FRESHLEY and S. GOLDSMITH, "Operating Experience With Plutonium Fuels in PRTR," AIME Symposium on Plutonium Fuels Technology, Scottsdale, Arizona (October 1967).

3. F. G. DAWSON, M. D. FRESHLEY, R. C. LIIKALA, D. L. PREZBINDOWSKI, V. O. UOTINEN, and H. H. VAN TUYL, "Results From USAEC Plutonium Utilization Program Conducted by Battelle-Northwest," BNWL-SA-3865, IAEA Panel on Plutonium Recycling in Thermal Power Reactors, Vienna, Austria (June 21-25, 1971).

4. T. B. BURLEY and M. D. FRESHLEY, "Internal Gas Pressure Behavior in Mixed-Oxide Fuel Rods During Irradiation," *Nucl. Appl. Technol.*, 9, 33 (1970).

5. J. K. BAHL and M. D. FRESHLEY, "Plutonium and Fission Product Migration in Mixed-Oxide Fuels During Irradiation," *Trans. Am. Nucl. Soc.*, 13, 599 (1970).

6. M. D. FRESHLEY and F. E. PANISKO, "The Irradiation Behavior of UO_2-PuO_2 Fuels in PRTR," BNWL-366, Battelle Northwest Laboratory (March 1967).

7. M. D. FRESHLEY, E. A. AITKEN, D. C. WADEKAMPER, R. L. JOHNSON, and W. G. LUSSIE, "Behavior of Discrete PuO_2 Particles in Thermal Recycle Fuel During Rapid Power Transients," *Trans. Am. Nucl. Soc.*, 13, 552 (1970).

ANS TRANS., vol. 14, 1971

41324 BEHAVIOR OF DISCRETE PLUTONIUM-DIOXIDE PARTICLES IN MIXED-OXIDE FUEL DURING RAPID POWER TRANSIENTS. Freshley, M. D. (Battelle-Northwest Labs., Richland, Wash.); Aitken, E. A.; Wadekamper, D. C.; Johnson, R. L.; Lussie, W. G. Nucl. Technol.; 15: No. 4, 239-48(Aug 1972).

Transient tests were conducted on unirradiated oxide pellet-containing fuel pins at SPERT to investigate the possible effects of large single 550 μm diameter PuO_2 particles on transient behavior. Results show that the effect of the large PuO_2 particles was to reduce the cladding failure threshold energy from the range of 225 to 274 cal/gm of fuel to the range of 200 to 213 cal/gm of fuel. Clad perforation occurs by localized melting caused by the expulsion of PuO_2 particles through the cladding. The presence of single 550 μm diameter PuO_2 particles in mixed-oxide fuels does not appear to affect significantly the cladding failure threshold energy from that of mixed-oxide fuels with the normal PuO_2 particle size and distribution. Therefore, product specifications which limit the maximum PuO_2 particle size to 550 μm diameter in mixed-oxide fuels do not appear warranted from the standpoint of transient fuel performance considerations. (auth)

NSA, vol. 26, 1972

4. The EEI Fuel Densification Program, *M. D. Freshley, S. Goldsmith (Battelle-Northwest)*

Irradiation-induced shrinkage or densification of uranium dioxide fuel pellets contributes to fuel column gapping and cladding collapse in light-water power reactor fuels. A program sponsored by the Edison Electric Institute (EEI) and 11 industrial sponsors[1] from the U.S., Europe, and Japan is being conducted by Battelle-Northwest Laboratories to investigate the principal causes of in-reactor fuel densification and relate these causes to fuel pellet characteristics and irradiation conditions.

As part of the program, nine distinct types of sintered UO_2 fuel pellets were produced with "tailored" structures to evaluate the effects of linear heat rating, burnup, and exposure time on densification. Several pellet types were also provided for evaluation by the participants in the program. The fuel types were categorized according to density (90 to 96% TD) and microstructural characteristics. Microstructural parameters include grain size, average pore size, pore size distribution, relative amounts of open and closed porosity, and pore shape. Average pore sizes range from 0.5- to 30-μm diam.

TABLE I

Irradiation Conditions for the Four EEI Fuel Densification Program Capsules

Item	Capsule A	Capsule B	Capsule C	Capsule D
Peak linear heat rating (thermal fission) kW/ft	9-10	9-10	9-10	9-10
Maximum fuel temp °C	1250-1375	1250-1375	1140-1450	1140-1450
Maximum fuel surface temperature, °C	~550	~550	~550	~550
Average cladding temp (peak location) °C	~335	~335	~335	~335
Coolant (NaK) temp (peak location) °C	~325	~325	~325	~325
Peak burnup, MWd/MTM	975-1040	3540-3790	3450-3790	3450-3790
Irradiation time (EFPH)	450	1630	1630	1630
Pin pressure (room-temp beginning-of-life), atm	1	1	1	53

A total of 326 pellet specimens representing 20 fuel types were irradiated in four 36-in.-long fuel pins contained in NaK-filled capsules. One of the pins (Pin C) included an axial restraint segment to simulate the effect of fuel cladding interaction on densification and one of the pins (Pin D) was pressurized to evaluate the effect of pre-pressurization on fuel densification. The irradiation conditions, which are typical of those for power reactor fuel, are summarized in Table I.

A major portion of the program effort is being devoted to the pre- and postirradiation characterization of the fuel pellets investigated. Pellet characterization included:

1. thermal studies—to determine thermal stability of the various fuel types included in the program by subjecting them to various ex-reactor thermal treatments
2. pre- and postirradiation density measurement—to determine changes in individual pellet density resulting from irradiation
3. detailed pre- and postirradiation microstructural analysis—to determine changes in pore size, pore size distribution, and grain size resulting from irradiation.

The extensive microstructural characterization utilizes optical and scanning electron microscopy techniques to quantify the effects of irradiation on pores as small as 0.1-μm diam.

In addition to providing data that will relate in-reactor fuel densification to fuel characteristics and irradiation conditions, this program is expected to provide information that will assist fuel suppliers in designing and producing sintered UO_2 pellet fuel that is stable during irradiation.

---

1. In addition to the Edison Electric Institute, sponsors are AGIP-Nucleare, ASEA-ATOM, Babcock and Wilcox, Central Research Institute of the Electric Power Industry (Japan), Combustion Engineering/Kraftwerk Union Aktiengesellschaft, Commissariat a l'Energie Atomique, Exxon Nuclear Company, Framatome, General Electric Company, Gulf United Nuclear Fuels Corporation/British Nuclear Fuels Limited, and the Japanese Atomic Industry Group.

ANS TRANS., vol. 18, 1974

15668 (CONF-680614–, pp 289-96) ELECTRON MICROSCOPE-MICROPROBE ANALYSIS OF PRECIPITATION. Gehlbach, R. E. (Oak Ridge National Lab., Tenn.). Dec 1971.
From twenty-second AEC metallography group meeting; San Diego, Calif. (19 Jun 1968).
Understanding the role of microstructure on mechanical and physical properties of alloy systems necessitates the identification of precipitated phases occurring in these materials. An electron probe microanalyzer accessory to the electron microscope, used in conjunction with standard electron microscope specimens, provides the means for obtaining a quick and reliable analysis of individual precipitates. The uniqueness of this tool is most apparent when using extraction replicas, permitting chemical analysis without matrix interference along with usual electron microscopy and diffraction of individual particles in the same spatial distribution as the bulk material, of the thin dendritic morphology often present in grain boundaries, of a size or distribution too fine to be resolved by conventional microprobe techniques, and in quantities not detectable by x-ray diffraction analysis of extraction residues. (T.F.D.)

NSA, vol. 26, 1972

28737 (NEDM-10735(Supp.6,7 and 8)) FUEL DENSIFICATION EFFECTS ON GENERAL ELECTRIC BOILING WATER REACTOR FUEL. (General Electric Co., San Jose, Calif. (USA). Atomic Power Equipment Dept.). Aug 1973. 260p.

The assumed effects of fuel densification considered are the potential for, (1) local power spikes resulting from axial fuel column gaps, (2) increased linear heat generation rate due to pellet axial shrinkage, (3) cladding collapse at the location of axial fuel column gaps, and (4) reduced pellet-clad thermal conductance due to increased pellet-to-clad gap as it may influence stored energy. The results of conservative analyses of local power spikes are presented. These analyses yielded the conclusion that there is >95% confidence that no more than one rod in any GE/BWR fuel type will have a power spike >5% in magnitude. This maximum spike magnitude will not occur at the limiting or highest power generation axial location in the rod. This analysis involves application of a maximum gap size which is larger than observed in BWR fuel. If the GE recommended maximum gap size were employed the resulting penalty is essentially zero. The results of the analysis of linear heat generation rate change due to densification show that the pellet axial shrinkage due to densification will be more than offset by the effects of axial thermal expansion. Thus no effect on LHGR is expected. The results of analyses of cladding creep collapse for existing BWR fuel types with more than one cycle of operation are also summarized. The results show that creep collapse will not occur. Evaluations were calculated for existing fuel operating through September of 1974. The gap conductances resulting from application of the AEC Staff model are presented for two alternative approaches to statistically treating the GE gap conductance data. These results are employed in the loss of coolant accident (LOCA) analyses for each fuel type. (auth)

NSA, vol. 29, 1974

15573 (HEDL-SA-270) FABRICATION MICROSTRUCTURE AND THERMAL CONDUCTIVITY IN UO_2–25 WT.PERCENT PuO_2 SINTERED PELLETS. Gibby, R. L.; Lawrence, L. A. (Hanford Engineering Development Lab., Richland, Wash.). Oct 1971. Contract AT(45-1)-2170. 22p. (CONF-711030–3). Dep. NTIS.

From twenty-fourth Pacific Coast regional meeting of the American Ceramic Society; Anaheim, Calif. (30 Oct 1971).

The results of a study are presented which show that thermal conductivity in this system can be significantly affected by the fabrication process. High-pressure-preslugging resulted in sintered mixed-oxide fuels with thermal conductivities lower than those predicted by a thermal conductivity-porosity relationship used for thermal performance analysis. Single-pressed fuels are also varied considerably in microstructure and thermal conductivity with values which ranged from good agreement with the reported thermal conductivity expression to 40% less. Thermal conductivities decreased linearly with increases in the laminar pore density. The pore shape factor β from the modified Maxwell-Eucken relationship increased linearily with laminar pore density from a value of approximately 1 for essentially uniform, isometric pore morphology, to 11 for the observed maximum estimated laminar pore density. The effects of structure on thermal conductivity were confirmed with melting heat rating experiments using single-pressed and high-pressure preslugged fuels. A minimum of 3 to 6% decrease in melting heat rating was observed for a 30% decrease in thermal conductivity due to the presence of laminar, interconnected pores instead of closed, isometric pores. (auth)

NSA, vol. 26, 1972

13060 IN-REACTOR MEASUREMENTS OF FUEL STACK SHORTENING. Hanevik, A. (OECD, Halden, Norway); Knudsen, K. D.; Arnesen, P. pp 89.1-89.4 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004-.

In-reactor measurements of UO_2 fuel stack shortening have been performed on a series of fuel rods with the following variables: pellet structural stability, fuel density, heat rating, and fuel to cladding clearance. The results indicate that the amount of stack shortening is primarily dependent on pellet structural stability. The rate of length change decreases with time (burnup), but stack shortening continues at a burnup of 4000 MWd/t. The effects of heat rating are negligible, and the length change is mainly dependent on burnup (number of fissions). It can be concluded from the measurements that the in-reactor stack shortening can be minimized by modifying pellet fabrication routes to give stable fuel structures. (auth)

NSA, vol. 32, 1975

2098 (BNWL-1778) GAPCON-THERMAL-1: A COMPUTER PROGRAM FOR CALCULATING THE GAP CONDUCTANCE IN OXIDE FUEL PINS. Hann, C. R.; Beyer, C. E.; Parchen, L. J. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)). Sep 1973. Contract AT(45-1)-1830. 235p. Dep. NTIS $7.60.

GAPCON-THERMAL-1, a modification of GAPCON, can be used to calculate the gap conductance and fuel temperatures in oxide fuel pins. The code in its current form was developed for the Regulatory Staff who use the code as a tool to independently calculate gap conductances for understanding various thermal performance models supplied by fuel venders. The code is capable of calculating fuel temperatures for several coolant, cladding, and fuel material combinations. Changes in the diametral gap width are modeled. The source of these changes include differential thermal expansion of pellet and cladding, elastic and creep deformation of cladding, fission product expansion of the pellet, and fuel expansion induced by cracks and subsequent thermal ratcheting. In addition to the gap changes, the code simulates a variety of fill gas compositions and changes to the gas composition caused by the release of fission gap and volatile impurities. Comparisons of calculations and experimental data indicate fuel temperatures are predicted reasonably well for short term irradiations and small gaps. The disagreement between predictions and experimental data is roughly proportional to the gap width and the calculated values are typically higher than the experimental values. Fuel temperature calculations for extended periods of irradiation become more uncertain because of inadequacies in the code and the dearth of well characterized data. These inadequacies are related principally to the kinetics of gap closure and fission gas release. On-going work is focused on reducing the uncertainty in these two areas of the code and in other areas of somewhat lesser significance. The effects of these uncertainties on the fuel stored energy are offset somewhat by the reduction in rod powers with burnup. (auth)

NSA, vol. 29, 1974

IN-PILE Ddensification OF URANIUM DIOXIDE

S.D.HARKNESS, S.R.PATI, M.G.ANDREWS, and W.P.CHERNOCK

Combustion Engineering, Inc.

SUMMARY

Irradiation of a wide variety of fuel types in 5 different irradiation facilities has demonstrated that stable fuel can be routinely fabricated. The roles of pore volume, pore size, and fuel grain size in controlling in-reactor densification are discussed in terms of a fission spike overlap model. Stable fuels with initial densities as low as 91% TD were fabricated. Evidence for a low temperature, 430°C (805°F), threshold for densification was discovered. Above this threshold densification rates increased slowly with temperature up to about 750°C (1382°F) upon which densification rates became nearly athermal. Internal fuel rod pressures at temperatures up to 0.84 kg/mm^2 (1180 psi) did not affect the densification characteristics of either stable or unstable fuels.

NUC. EN. MATUR., vol. 3, pt. 1, 1976

53547 (AREAEE-135) DENSIFICATION OF URANIUM DIOXIDE: A LITERATURE REVIEW. Haroun, N. A. (Atomic Energy Establishment, Cairo (Egypt). Dept. of Metallurgy). 1972. 27p. Dep. NTIS (U. S. Sales Only).

Factors affecting the densification of UO_2 are extensively reviewed. It was concluded that low-temperature sintering of nonstoichiometric oxides in neutral atmospheres is most promising. Variables discussed include, process variables, physical and physicochemical variables, and chemical variables. (101 references). (auth)

NSA, vol. 26, 1972

26276 VOLUME CHANGES IN IRRADIATED UO_2 FUEL. Hastings, I. J. (Atomic Energy of Canada Ltd., Chalk River, Ont.); Rose, D. H.; Schankula, M. H. J. Amer. Ceram. Soc.; 58: No. 1-2, 74(1975).

This note reports volume changes in UO_2 fuel irradiated in WRI, an organic-cooled, heavy-water moderated research reactor located at Pinawa, Manitoba. Four UO_2 fuel elements, clad with Zr-2.5 wt % Nb and irradiated in the reactor under similar conditions, were selected for examination. Samples 1 by 1 by 5 mm were cut from radial positions in the sections; their density was measured by an immersion technique used successfully with U_3Si fuel. The precision cutting technique permitted accurate determination of the positions of the samples in the fuel. Volume changes were correlated with microstructure observed by optical and electron microscopy. Regions of irradiation-induced volume change identified in UO_2 fuel occurred: (1) at $\lesssim$1300°K, where densification occurs by removal of sintering porosity; (2) from 1300 to 1900°K, where swelling caused by intergranular fission gas bubbles predominates; and (3) at >1900°K, in the region of columnar growth, where both swelling and densification are observed. (PMA)

NSA, vol. 31, 1975

25391 (WCAP-8219-A) FUEL DENSIFICATION EXPERIMENTAL RESULTS AND MODEL FOR REACTOR APPLICATION. Hellman, J. M. (ed.). (Westinghouse Electric Corp., Pittsburgh, Pa. (USA). Nuclear Energy Systems Div.). Mar 1975. Westinghouse Electric Corp., Pittsburgh, PA.

Westinghouse has developed new Fuel Densification and Power Spike Models and some changes for DNB evaluation based on the performance of fuel in operating reactors and associated experimental results. The data input for these models included measurements of fuel stack height shrinkage and interpellet gap distributions from incore flux traces, gamma scanning and destructive testing of irradiated fuel rods, incore measurements of neutron flux spikes from interpellet gaps of known size, and DNB testing of simulated fuel rods with axial power spikes. The experimental instrumentation, data collection and analysis, and the derivation and application of the models are described. (auth)

NSA, vol. 32, 1975

3. In-Reactor Densification Experience with Gulf United Fuel, *R. B. Holden, N. Fuhrman (GUNFC), L. Raven (BNFL)*

Since 1968 more than 20,000 Zircaloy-clad UO_2 fuel rods fabricated by Gulf United have been irradiated in both BWRs and PWRS. No collapsed cladding nor any indication of significant pellet column gaps have been found on any of this fuel, which was fabricated to a nominal density of 93.5% of theoretical. Nevertheless, the microstructure of the standard Gulf United production fuel made to date was believed to be susceptible to in-reactor densification and, for this reason, a method for producing UO_2 having a controlled porosity microstructure, predicted to be resistant to densification, has been developed.

To produce UO_2 fuel which has both thermal and irradiation stability, it is necessary to substantially eliminate small porosity and have most of the pore diameters be $>5\mu$. In collaboration with BNFL, a process was developed that is capable of producing such a controlled porosity microstructure. Extensive evaluation of controlled porosity pellets has been carried out and fuel production methods have been developed and operated.

Twelve full-sized water reactor fuel rods, each having a fuel column length of 144 in., were fabricated with well-characterized controlled porosity pellets and irradiated in two assemblies in the SGHWR. The fuel has a mean pre-irradiation density of 92.0% of theoretical and the assemblies were discharged after 116 and 186 equivalent full power days of operation, respectively. The mean density observed among pellets exposed to a range of linear power ratings from 5.5 to 14 kW/ft was 92.3%, an increase of only 0.3%; thus fully verifying resistance of this fuel to in-reactor densification.

Comparison of the pellet microstructure before and after irradiation confirms that at temperatures below 1300°C, micropores (<2-μ diam) originally present tend to disappear, while the larger controlled pores are stable. In the higher temperature regions even the controlled pores undergo a volume reduction; however, the porosity decrease appears to be balanced by the generation of a substantial amount of observable intergranular fission gas bubble porosity,

Similarly produced fuel is undergoing irradiation in several Yankee Rowe Core X rods. In addition, three Gulf United qualification assemblies now in their second cycle of irradiation in the Connecticut Yankee reactor contain UO_2 pellets, made by a generically related process, that exhibit some of the large pore features of the controlled pore microstructure. In-core flux monitor traces in the Connecticut Yankee assemblies have not shown any flux blips indicative of pellet column gap formation.

ANS TRANS., vol. 17, 1973

30852 (AAEC/E-220) INFLUENCE OF PRECIPITATION CONDITIONS ON THE PROPERTIES OF AMMONIUM DIURANATE AND URANIUM DIOXIDE POWDERS. Janov, J.; Alfredson, P. G.; Vilkaitis, V. K. (Australian Atomic Energy Commission Research Establishment, Lucas Heights). May 1971. 40p. Dep. NTIS (U. S. Sales Only).

Factors affecting the properties of ammonium diuranate (ADU) precipitates in relation to the properties of the subsequent UO_2 powders in pellet fabrication are reported and the importance of precipitation parameters demonstrated. Variables investigated include continuous single-versus two-stage precipitation, pH, residence time, washing of ADU to remove nitrate, and calcination-reduction conditions. The most important variable was the pH at which precipitation occurred. This governed the size of agglomerates, which determined the settling and filtering characteristics of the ADU slurry. In two-stage precipitation, the ADU properties were determined by the proportion of uranium precipitated at different pH values. Washing of nitrate from ADU appeared not to affect the properties of the subsequent UO_2 powder. Approximately 75% of the initial nitrate was removed by washing once in demineralized water; more extensive washing caused only a slight further reduction in nitrate content and significantly reduced the filterability of the ADU slurry. When ADU was reduced to UO_2 at 600°C in hydrogen, differences in the surface areas of the powders were markedly reduced but the agglomerate structure of the ADU was retained. ADU that was precipitated mainly at pH 3 to 4, and contained large agglomerates, gave UO_2 containing large agglomerates, which sintered poorly and the pseudomorphs of the large agglomerates were still discernible in the sintered pellets. ADU precipitated at high pH contained small agglomerates and gave UO_2 powder that sintered readily to high-density pellets with uniform microstructure. The settling rate of the ADU slurry gave an early indication of the likely sinterability of the resultant UO_2 powder since both were functions of agglomerate size. Reasonably filterable ADU and sinterable UO_2 powders were prepared via single-stage precipitation at pH 7.2 or by two-stage precipitation in which less than 80% of the uranium was precipitated in the first stage at pH 3.5 and the remainder in the pH range 7 to 8. Less stringent control was required in two-stage precipitation than in single-stage precipitation, but this was offset by the additional complexity of two precipitators and their auxiliaries. (auth)

NSA, vol. 26, 1972

31390 (ANL-7901) THERMODYNAMIC AND DEFECT-STRUCTURE STUDIES IN MIXED-OXIDE FUELS. Javed, N. A.; Roberts, J. T. A. (Argonne National Lab., Ill.). Feb 1972. Contract W-31-109-eng-38. 22p. Dep. NTIS.

Transpiration experiments were performed at 1000 to 1700°C to study the thermodynamics and defect structure of hypostoichiometric UO_2 – 20 wt % PuO_2. The oxygen partial pressures were established by using flowing H_2/H_2O mixtures. After equilibration, the quenched products were analyzed by chemical, x-ray, neutron-diffraction, and metallographic techniques. The principal thermodynamic functions $\Delta\bar{G}_{O_2}$, $\Delta\bar{H}_{O_2}$, and $\Delta\bar{S}_{O_2}$ were calculated between 1400 and 1700°C for various oxygen-to-metal ratios. Based on x-ray, neutron-diffraction, and metallographic data, it was concluded that the 20 wt % mixed oxide exists as a single phase under normal conditions, even at an oxygen-to-metal ratio as low as 1.92. The data from density measurements and neutron-diffraction analysis indicated that the predominant defects in the hypostoichiometric UO_2 – 20 wt % PuO_2 are anion vacancies. The curves of $\Delta\bar{G}_{O_2}$ versus temperature for the mixed oxide with different oxygen-to-metal ratios were compared with $\Delta\bar{G}_{O_2}$ curves for various metal oxides, and possible interactions among fuel, cladding, fission product, and coolant were established. (auth)

NSA, vol. 26, 1972

A General Electric Fuel Performance Update,

Frank D. Judge, Harold E. Williamson, David T. Weiss (GE–USA)

I. INTRODUCTION

Over the past 15 years, General Electric has amassed a large fuel performance data base consisting of over 500,000 production fuel rods in over 10,000 fuel bundles, operating in nearly 30 boiling water reactors. We also recognize that reliable fuel performance is a necessity for achieving high plant availability and low energy costs. In the continuing quest to provide the best possible fuel, we have maintained an aggressive fuel surveillance program. Because of this program, when fuel problems are encountered we are able to effectively search out the affected fuel, determine the cause of the problem, and take corrective action.

In this paper we will attempt to summarize some of the facets of fuel and fuel channel performance we believe to be of current interest.

II. IT'S A NUMBERS GAME

With nuclear fuel, only a very few fuel failures can be tolerated and still achieve a high plant availability. A fuel rod failure rate in the range of 0.1 to 1.0% is undesirable. To demonstrate adequate performance throughout the fuel lifetime requires operating large numbers of fuel rods to both high power (kW/ft) and burnup (MWd/MT). For example, to expose these low-frequency failure mechanisms requires operating between 1000 and 10,000 rods in the power and exposure regime of interest.

The defeat of a low probability mechanism can become a very frustrating statistical battle. In recent years, the rod moisture problem was an example where many thousands of data points were required to accurately establish the rod moisture distribution. It became necessary to eliminate a very small "statistical tail" to solve the problem.

We believe our vast data base of more than 500,000 fuel rods in service is an invaluable fuel performance resource.

III. SURVEILLANCE

Just having a large experience base is not enough; we believe it is very important to know what is happening in detail. To this end, GE has an aggressive fuel and channel surveillance and inspection program. Particular emphasis is put on the lead fuel and channels of each unique configuration (e.g., improved 7×7, 8×8).

It is not possible, of course, to examine all fuel at all plants with nearly 30 GE BWRs now in operation. But it is possible, by combining a lead bundle program with a program responsive to individual plant needs, to plan an effective surveillance program. A problem can be anticipated by monitoring plant operating data, especially the plant off-gas trends.

The GE fuel bundle is compatible with a fuel surveillance program in that an irradiated bundle may be readily disassembled underwater to permit individual rod removal and examination—and then be reassembled for further irradiation. By examining individual rods using nondestructive testing (NDT) and visual techniques, the ability to locate a defected rod is increased by a factor of about 50 over inspection techniques where the bundle periphery is inspected visually by television. To date, more than 37,000 individual production fuel rods have undergone nondestructive examination. Rods of specific interest can and have been transferred to the hot cells, such as the Vallecitos Nuclear Center in the USA, Grosswelzhein in Germany, or Kjeller in Norway for additional evaluations.

The channel surveillance program involves the combination of visual examination and dimensional measurements. To date, more than 100 different channels have been measured to 7 GE-BWRs; most of these have been given repeat measurements at later outages.

IV. RECENT FUEL EXPERIENCE

We have recently been through the difficult, and well-documented period of moisture- or hydride-related fuel rod failures. The cause of the problem was determined to be excessive moisture in fuel rods and action was taken to eliminate the problem. The solution being ultimately and predominately high-temperature vacuum outgassing of loaded fuel rods prior to final seal welding, and the addition of a hydrogen getter in the fuel rod to provide added assurance that no hydride-related failures will occur. That the problem has been eliminated is well demonstrated. First, by the excellent performance of 325 reload fuel bundles of the "improved"[a] 7×7 design which are loaded in 7 different GE-BWRs. These bundles have been tested by sipping after one cycle of operation. In these 325 bundles, only one leaker rod is suspected—that's one rod out of 15,925. And finally, three new large GE-BWRs, also predominately with improved 7×7 fuel, are well into their first operating cycle with very satisfactory fuel performance noted to date, as indicated by very low plant off gas.

Another fuel performance problem that has received a great deal of attention lately is pellet-clad-interaction (PCI). Where hydride failures can be generally characterized as occurring early in life, usually in the 0 to 11,000 MWd/MT range, and tending to occur somewhat randomly in the core and in a bundle; PCI failures tend to occur later in life, in the range 6000 to 27,500 MWd/MT, and can be a function of exposure, power, power change, design characteristics, and manufacturing variables.

Based heavily on work performed at Halden, where GE is an active participant, and at our own General Electric Test Reactor (GETR) and development laboratories, the mechanism was reproduced and changes to minimize the likelihood of PCI occurrence were determined. Changes were specified for both the cladding and the pellet geometry.

---

[a]The term "improved" will be defined later in this section.

The cladding changes involved specifying a higher annealing temperature to achieve optimum uniformity of mechanical properties, and increasing cladding thickness to compensate for the lower yield strength of the new material. To minimize the potential for pellet ridging, a shorter, chamfered pellet with no dishing was specified. The first production fuel of this type was built during the second half of 1972 and is herein referred to as "improved" 7 × 7 fuel, to differentiate from earlier 7 × 7 fuel.

Because PCI is a statistical phenomenon occurring at high exposure, it will require the combination of large numbers of fuel rods operated to high exposure to demonstrate the effectiveness of the PCI fix. It is expected that the first "improved" 7 × 7 reload fuel bundles, put into service during the Spring of 1973 outages, will begin to provide meaningful data on PCI during their second cycle of operation. Hence, the tests and examinations performed during the Spring of 1975 outages will be of great interest. In the meantime, we are continuing to test fuel in Halden and GETR and are developing sophisticated analytical models to increase our understanding of the PCI mechanism.

With the introduction of BWR-6, GE is specifying an 8 × 8 fuel bundle. Since PCI failures decrease with decreasing fuel linear heat rate (kW/ft), it was determined to retrofit 8 × 8 fuel as reload fuel into operating plants and into not-yet-fabricated initial cores (BWR 4 and 5) to assure a high degree of fuel reliability and also to provide additional margin to thermal and thermal-hydraulic limits. The first 8 × 8 fuel was loaded, as reload fuel, during 1974. Consistent with the schedule for data feedback on "improved" 7 × 7 fuel, approximately two years will be required before statistically significant data will be obtained on this "improved" 8 × 8 fuel.

The benefit to fuel performance of operation at low linear heat rates has been well demonstrated on some of the early BWR-1s. A good example is the Garigliano initial core, which has operated to an average exposure of over 15,000 MWd/MT with only four leaker fuel bundles.

V. PLUTONIUM RECYCLE FUEL

General Electric has been involved with two significant development programs oriented toward providing a sound experience base for the orderly entry into the plutonium recycle fuel business. First, GE has been a participant in the Ente Nazionale per L'Energia Elettrica-sponsored plutonium development program in the Garigliano reactor. Second, we have participated in the Edison Electric Institute (EEI)-sponsored program, designing and building mixed-oxide fuel rods and bundles for irradiation in the Big Rock Point reactor.

We are now in the final stages of producing a fuel reload batch of mixed-oxide fuel bundle for loading in Garigliano during 1975. This batch of fuel will incorporate over 1400 mixed-oxide fuel rods. These rods were fabricated at Belgonucleaire, with uranium rod fabrication and bundle assembly taking place at Fabbricazioni Nucleari in Italy.

VI. FUEL CHANNELS

GE now has in service approximately 9000 Zircaloy fuel channels. As mentioned in Sec. III, the performance of these channels is being closely followed. Much useful data have been obtained. Using channel measurement data obtained from the production channels, we have established a channel creep model for use in predicting channel behavior. Of particular interest to BWR-6 owners is the performance of four pre-production 3.0-mm-thick channels now being irradiated in KKM. These channels, first loaded in January 1974, were given an interim inspection during the plants' August 1974 outage and were found to be performing in a completely satisfactory manner.

VII. CONCLUDING STATEMENT

General Electric has a large and growing boiling water reactor experience base with Zircaloy-clad pellet fuel. Also important, we have an aggressive surveillance program to permit us to obtain meaningful data out of this experience base. We have taken actions to eliminate, or at least severely restrict, the consequences of hydride and PCI failures. Of great significance is the fact that current performance requirements placed on the fuel are generally within the experience of statistically demonstrated fuel capability. This gives us a high degree of confidence that no new fuel problems will be encountered.

ANS TRANS., vol. 20, 1975

259882 Koizumi, Masumichi; Furuya, Hirotaka; Yokouchi, Yoji; Kajitani, Mikio; Yamaguchi, Toshihiro (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). **Fuel restructuring and gap conductance of thermal reactor fuels.** Under the irradiation in JRR-2.

Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan). Semi-annual progress report of Power Reactor and Nuclear Fuel Development Corporation, Tokai Works. Jan.-Jun., 1974. **PNCT-831-74-02.** Dec 1974. p. 84-88.

Fuel restructuring, that is central void and columnar and equiaxial grain growth, was observed by ceramography, and the gap conductance between fuel and cladding under irradiation was measured. UO_2 pellets, approximately 12.5 mm diameter and 10 mm high, were irradiated to observe the restructuring in three fuels of different density, and to measure gap conductance which was measured as a function of linear heat rate. One or two pins were set together in an aluminum capsule with a thermocouple for each fuel pin. In the gap conductance test, an additional thermocouple fabricated from W-WRe and sheathed with tantalum was set at the radial center of fuel. The post-irradiation examination was conducted in the hot laboratory of Japan Atomic Energy Institute. In the lower density fuel, the fuel restructuring is drastic in spite of comparatively lower linear heat rating. In the equiaxial grain growth region, a number of radially elongated voids were observed, which separated columnar and undisturbed grain growth regions clearly. On the other hand, in the high density fuel, fuel structure is smoothly continuous from columnar grain growth region to undisturbed grain growth region. Gap conductance was calculated from the temperature readings only during the time of reactor start-up. It is thought that the gap of 0.05 mm diameter was closed, and fuel and cladding came to direct contact at this point. (Iwakiri, K.).

FUEL PINS: physical radiation effects; URANIUM DIOXIDE: fuel pellets.

Atomindex, vol. 7, 1976

31434 MICROSTRUCTURE OF SOL-GEL-DERIVED $(U,Pu)O_2$ MICROSPHERES AND PELLETS. Lackey, W. J.; Bradley, R. A. (Oak Ridge National Lab., Tenn.). Nucl. Technol.; 14: No. 3, 257-68(Jun 1972).

The microstructure of sintered $(U,Pu)O_2$ microspheres and pellets prepared by the sol-gel process was characterized using light and electron microscopy, electron microprobe analysis, and alpha autoradiography. The material is typically homogeneous in composition, and the grain size is usually between 1 and 5 μm. Coarse microspheres have densities greater than 95% of theoretical, but the fine microspheres approach theoretical density. The microstructure of low-density pellets consists of high-density and medium-density regions in a porous matrix. (auth)

NSA, vol. 26, 1972

45930 (HEDL-TME-72-81) EFFECTS OF PORE STRUCTURE ON THE MELTING HEAT RATING OF OXIDE FUELS. Lawrence, L. A.; Gibby, R. L. (Hanford Engineering Development Lab., Richland, Wash.). May 1972. Contract AT(45-1)-2170. 41p. Dep. NTIS.

Mixed uranium–plutonium oxide fuels prepared by different fabrication techniques can have significantly different thermal properties. These differences, which stem from differences in pore shape and distribution characteristics resulting from the fabrication process, were investigated using thermal diffusivity techniques and in-reactor experiments. Fuel characterized by large amounts of linear porosity was found to have a thermal conductivity 30 to 40% lower than fuel of equivalent density with uniformly distributed fine spherical porosity and large isometric pores. Results of the irradiations confirmed a lower melting heat rating for fuel characterized by large amounts of linear porosity. Results of the irradiations were compared to temperature profile predictions based on the laboratory thermal diffusivity data. Reasonable agreement is obtained if expected effects of oxygen redistribution down the thermal gradient in the fuel are considered. (auth)

NSA, vol. 26, 1972

40411 LATTICES OF PLUTONIUM-ENRICHED RODS IN LIGHT WATER. II. THEORETICAL ANALYSIS OF PLUTONIUM-FUELED SYSTEMS. Liikala, R. C.; Uotinen, V. O.; Jenquin, U. P. (Battelle-Northwest Labs., Richland, Wash.). Nucl. Technol.; 15: No. 4, 272-96(Aug 1972).

A theoretical analysis of UO_2–PuO_2 fueled, light water moderated lattice experiments was performed to aid in establishing technical bases and design criteria for the utilization of Pu bearing fuel in thermal power reactors. Results for UO_2 and Al–Pu lattices are included to understand the effects due to U and Pu separately. The problems involved in calculating critical experiments are discussed. Estimates of the effects of various approximations inherent in the theoretical methods and/or analysis procedures are included along with the consequence on the results of the correlation. Uncertainties in the measurements and the neutron cross-section data are related to uncertainties in the calculated values of k_{eff}. Areas which should be investigated in future analyses are also identified. (auth)

NSA, vol. 26, 1972

30124 MODEL FOR IN-REACTOR DENSIFICATION OF UO_2. MacEwan, S. R.; Hastings, I. J. (Atomic Energy of Canada Ltd., Chalk River, Ont.). Phil. Mag.; 31: No. 1, 135-143(Jan 1975).

A model for the in-reactor densification of UO_2 at temperatures below about 1200°K was developed. The approach is based on the buildup of irradiation-produced point defects during a transient period and their annealing at sintering pores. At 900°K and 10^{19} fissions m^{-3} s^{-1}, the model shows that intragranular pores up to 0.2 μm diameter disappear in about 1 hour; those above 1 μm diameter take about 75 days. Agreement with existing results is reasonable. (auth)

NSA, vol. 31, 1975

5707 (NEDO-12440) IN-REACTOR DENSIFICATION BEHAVIOR OF UO_2. Marlowe, M. O. (General Electric Co., Pleasanton, Calif. (USA). Vallecitos Nuclear Center). Jul 1973. 31p. General Electric Co., San Jose, CA 95114.

The processes leading to the densification of porous nuclear fuels during irradiation are reviewed. It is concluded irradiation-induced diffusion plays a dominant role, i.e., the kinetics of densification can be no faster than diffusion will allow. Models for densification controlled by volume diffusion are modified to use irradiation-induced diffusion as the material transport mechanism. An experiment to determine the irradiation-induced diffusion in uranium dioxide is reviewed. The model and measured diffusivity are applied to the prediction of the densification behavior of uranium dioxide based on typical grain growth and densification rates. The heat-treatment conditions (time-temperature) for thermal simulation of a specific irradiation exposure in terms of fuel densification are estimated. Specific microstructural features leading to densification resistance are discussed. (auth)

NSA, vol. 32, 1975

24611 CALCULATIONAL PROCEDURE FOR DETERMINING CREEP COLLAPSE OF LWR FUEL RODS. Merckx, K. R. (Exxon Nuclear Co., Richland, Wash. (USA)). Nucl. Eng. Des.; 31: No. 1, 95-101(Nov 1974).

Observed collapses in pressurized water reactor fuel rods have been attributed to the radiation enhanced creep of Zircaloy cladding into regions where separations in the fuel pellet stack have occurred. A computer code, COLAPX, has been written to determine the growth of ovality and the ultimate collapse of fuel rod cladding under reactor operating conditions. The theoretical bases of this code, the finite element formulation used, the constitutive relations between the displacement fields and the element forces, and the radiation, temperature and stress dependent material model for creep of Zircaloy tubing are presented. Comparisons of the creep rate predictions and of the ovality predictions with data from irradiated tubes and fuel cladding are presented. (auth)

NSA, vol. 31, 1975

31432 USE OF MICROSTRUCTURAL DATA FOR VERIFICATION OF FUEL MATERIAL SWELLING MODELS. Merckx, K. R. (WADCO Corp., Richland, Wash.). Nucl. Eng. Des.; 18: No. 1, 41-51(1972). (CONF-710903-12).

From first international conference on structural mechanics in reactor technology; Berlin, Germany (20 Sep 1971).

Methods for verifying proposed models of reactor fuel material swelling are described using quantitative data obtained from microstructural observations. Because of the range and interaction of the mechanical, thermal, and radiation conditions encountered by reactor fuel elements, microscopic structural observations provide the only source that contains sufficient information to verify fuel swelling models. Various methods for quantitatively interpreting microstructural data are related to their uses for verifying swelling models. Methods for evaluating accumulative properties (volume increase, bubble density count, and specific bubble surface area) in fuel materials having random spatial distributions (describable by a Poisson process) are used to evaluate empirical models. New methods for estimating these properties are described. Size distributions of bubbles are predicted by some models: thus, methods for estimating such size distributions are used to verify these models. A new development for evaluating statistical estimations of size distributional parameters is described. The most detailed and advanced swelling models make predictions on both the size and spatial distributions of swelling bubbles. The combined probabilistic formulation of these models with specialized statistical estimation procedures provides the means for testing the hypotheses used in these models. An application of this technique to a grain boundary dependent swelling model is described. (auth)

NSA, vol. 26, 1972

12739 SINTERING OF UO_2: A STUDY OF THE DEVELOPMENT OF HIGH DENSITY AND CONTROLLED MICROSTRUCTURE IN THE SINTERED URANIA PELLETS. Moorthy, V. K.; Anandan, N. S. (BARC, Bombay). Trans. Indian Ceram. Soc.; 27: No. 5, 145-58(1968).

Studies were made on the relations between powder properties, sintering treatments, and densification and microstructure development in urania pellets. Urania powders were prepared from ammonium diuranate by decomposition and reduction at various temperatures, namely, 400, 600, 900, 1200, and 1500°C. Pellets prepared from these powders were sintered from 1200 to 1700°C with different soaking periods. The influence of sintering temperature, soaking periods at different temperatures, intermediate soaks at varying time periods at various temperatures, reheating of prepared sintered pellets, and rates of heating during sintering on the density and microstructure development of pellets was studied. The study revealed that urania pellets of specified density and grain development can be prepared by first attempting to achieve densification and then continuing the thermal treatments to attain the desired grain size and microstructure. (Indian Sci. Abstr.)

NSA, vol. 26, 1972

27428 STUDY OF DENSIFICATION OF NUCLEAR FUEL FOR LIGHT-WATER TYPE POWER REACTOR. Morishima, A.; Harayama, Y.; Uchida, M. (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment); Tobioka, T. Genshiryoku Kogyo; 20: No. 6, 19-25 (Jun 1974). (In Japanese).

Densification of fuel pellets was discovered in the Beznau Reactor of Switzerland in 1971. It is important to study this phenomenon from the viewpoint of reactor safety. The final density and axial contraction of fuel pellets were analyzed statistically by using experimental data on fuel irradiation. The increase of linear power density due to the densification of pellets was estimated, and the lowering of heat transfer of the gap and hang-up of pellets due to creep were also studied. The considered mechanisms of the densification were the mechanism in relation to resolution and the mechanism in relation to spike due to nuclear fission. Various experiments on densification are in progress. The experiment performed with the Halden reactor, Norway, is described. Contraction of all fuel rods were observed, and quantitative data on the contraction are presented. (JA)

NSA, vol. 32, 1975

40448 MEASUREMENT OF k_∞ AND RELATIVE REACTION RATES IN AN H_2O-MODERATED UO_2–PuO_2 PARTICULATE FUELED LATTICE. Newman, Darrell F. (Pacific Northwest Labs., Richland, Wash.). Nucl. Technol.; 15: No. 4, 192-208 (Aug 1972).

The neutron multiplication factor, k_∞, and relative reaction rates were measured at room temperature in the Physical Constants Testing Reactor for a water moderated lattice fueled with Zircaloy clad UO_2–PuO_2 fuel rods. The plutonium content in different sets of fuel rods ranged from 0.9 to 2.5 wt% PuO_2 in UO_2. These experiments provide neutronics data for uniform lattices of mixed oxide fuels which contain plutonium oxide distributed throughout the uranium oxide as finite-sized particles. The PuO_2 particle sizes in different sets of fuel rods ranged between zero and 328 microns in diameter. Knowledge of the effect that plutonium particle size has on neutron multiplication is important for the economic utilization of plutonium. An experiment-theory correlation provides a basis for assessing the adequacy of calculational techniques. (29 references) (auth)

NSA, vol. 26, 1972

6. Steady-State Creep Model for UO_2, *C. S. Olsen (ANC)*

An analytical expression for the steady-state creep of UO_2 is required for use in computer codes that predict fuel dimensional changes during irradiation. Out-of-pile[1-8] and in-pile[9-15] experiments have been reviewed and an analytical model based on theoretical models and experimental data for the steady-state creep of UO_2 has been developed and verified. The effects of temperature, stress, density, grain size, and fission rate on the various measured UO_2 creep rates were evaluated and are reflected in the model, represented by Eq. (1).

$$\dot{\epsilon} = \frac{(A_1 - A_2\dot{F})\sigma \exp(-Q_1/RT)}{(A_3 + D)G^2} + \frac{A_4\sigma^{4.5} \exp(-Q_2/RT)}{(A_5 + D)} + A_6\sigma\dot{F} \exp(-Q_3/RT) \quad , \qquad (1)$$

where

$A_1 = 9.728 \times 10^6$

$A_2 = 3.24 \times 10^{-12}$

$A_3 = -87.7$

$A_4 = 1.376 \times 10^{-6}$

$A_5 = -90.5$

$A_6 = 9.24 \times 10^{-28}$

$Q_1 = 90{,}000$ cal/mole

$Q_2 = 132{,}000$ cal/mole

$Q_3 = 5200$ cal/mole

$\dot{\epsilon}$ = creep rate $\frac{\text{in.}}{\text{in.-h}}$

$\dot{F}$ = fission rate [8.4×10^{17} to 1.18×10^{20} fission/(m^3-sec)]

σ = stress (1000 to 16,000 psi)

T = temperature (713 to 2073 K)

D = density (92 to 98% TD)

G = grain size (4 to 35 μm)

R = gas constant [1.987 cal/(g-mole-K)].

The first two terms of this equation reflect high-temperature thermal creep mechanisms. The first term represents a viscous creep mechanism with some grain boundary sliding occurring at low stresses. The viscous creep is linearly proportional to stress; inversely proportional to the square of grain size; and enhanced by a factor that is proportional to the fission rate. A dislocation-climb mechanism, expressed in the second term of Eq. (1), operates at stresses greater than a transition stress that is governed by the grain size.[5,16] The dislocation-climb creep rate is proportional to the stress to approximately the 4.5 power. Data are not available for irradiation enhancement of UO_2 creep in the dislocation-climb-controlled regime and, therefore, although certain authors[16] have suggested equations incorporating irradiation-enhanced dislocation-climb mechanisms, such mechanisms have not been incorporated into this model.

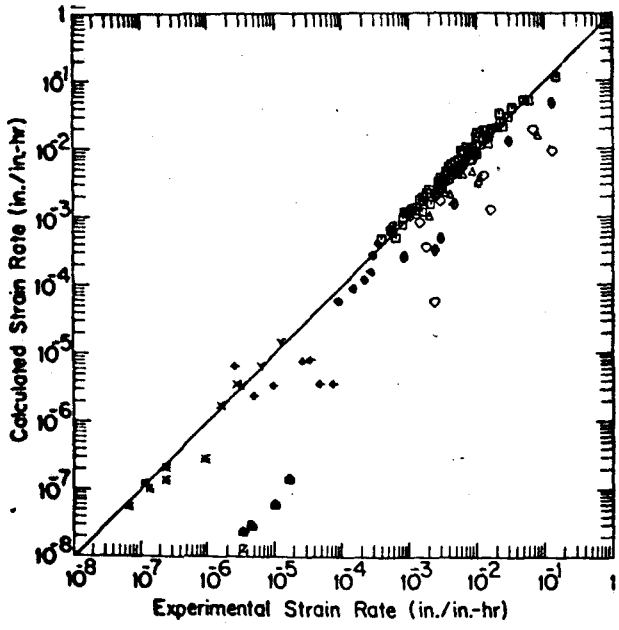


Fig. 1. Comparison of experimental data for irradiated and unirradiated UO_2 with corresponding calculated values obtained from Eq. (1).

- Bohaboy (6) - Unirradiated
- Poteat and Yust (4) - Low Stress Region, Unirradiated
- Poteat and Yust (4) - High Stress Region, Unirradiated
- Wolfe and Kaufman (5) - 18-μ Grain Size, Unirradiated
- Wolfe and Kaufman (5) - 55-μ Grain Size, Unirradiated
- Sykes and Sawbridge (9) - Irradiated
- Brucklacher and Dienst (11) - Irradiated
- Clough (10) - Irradiated
- Solomon and Gebner (12) - Irradiated
- Perrin (13) - Irradiated
- Perrin (13) - Unirradiated

The stress necessary for transition from viscous creep to dislocation creep is assumed to be independent of temperature because of the apparently small difference in measured activation energies for the two mechanisms. The transition stress (σ_{trans}) is given by the following expression:

$$\sigma_{trans} = 24{,}000G^{-0.5714} \quad . \qquad (2)$$

When the stress is less than σ_{trans}, the actual stress in the fuel is used in the first term of Eq. (1); for stresses greater than σ_{trans}, the transition stress is used in the first term of Eq. (1). For stresses below σ_{trans}, the contribution to the creep rate from the second term is negligible.

(Cont'd pg 104)

The fission process induces creep at low temperatures for which out-of-pile thermal creep normally does not occur. This fission-induced creep rate is represented by the last term in Eq. (1) and is proportional to stress and fission rate. Early experiments reported the process to be athermal,[14] but suggested that the temperature dependency could be masked by material variations. Later, Brucklacher[15] reported a slightly temperature-dependent process with an activation energy of 5200 cal/mole. The effect of material properties on the irradiation-induced creep of UO_2 has not been determined.

Analytical predictions obtained using Eq. (1) are compared in Fig. 1 with experimental data selected from compressive creep tests. Good agreement with all the data is obtained except with the data from Poteat[4] in the high stress region, data for the 55-μm grain size material used by Wolfe,[5] and data from Perrin[13] and Solomon et al.[12] for unirradiated fuel. No explanation is available for these discrepancies.

1. R. SCOTT et al., *J. Nucl. Mater.*, 1, 39 (1959).
2. W. M. ARMSTRONG et al., *J. Nucl. Mater.*, 7, 133 (1962).
3. W. M. ARMSTRONG et al., *J. Nucl. Mater.*, 9, 121 (1963).
4. L. E. POTEAT and C. S. YUST, "Grain Boundary Reactions During Deformation," ORNL-P-2371, Oak Ridge National Lab. (1966).
5. R. A. WOLFE and S. F. KAUFMANN, "Mechanical Properties of Oxide Fuels," WAPD-TM-587, Bettis Atomic Power Lab. (1967).
6. P. E. BOHABOY et al., "Compressive Creep Characteristics of Stoichiometric Uranium Dioxide," GEAP-10054, General Electric Co. (1969).
7. B. BURTON and G. L. REYNOLDS, *Acta. Met.*, 21, 1073 (1973).
8. B. BURTON and G. L. REYNOLDS, *Acta. Met.*, 21, 1641 (1973).
9. E. C. SYKES and P. T. SAWBRIDGE, "The Irradiation Creep of Uranium Dioxide," BC/BW/1489, Central Electricity Generating Board (1969).
10. D. J. CLOUGH, "Irradiation Induced Creep of Ceramic Fuels," *Fast Reactor Fuel and Fuel Elements*, p. 321, Gesellschaft fuer Kernforschung, Karlsruhe (1970).
11. D. BRUCKLACHER and W. DIENST, *J. Nucl. Mater.*, 36, 244 (1970).
12. A. A. SOLOMON and R. H. GEBNER, *Nucl. Technol.*, 13, 177 (1972).
13. J. S. PERRIN, *J. Nucl. Mater.*, 39, 175 (1971).
14. A. A. SOLOMON et al., "Fission-Induced Creep of UO_2 and its Significance to Fuel-Element Performance," ANL-7857, Argonne National Lab. (1971).
15. D. BRUCKLACHER et al., "Creep Behavior of Oxide Fuels Under Neutron Irradiation," translated from German by J. L. ROUTBORT, ANL-Trans-942, Argonne National Lab. (1973).
16. M. S. SELTZER et al., *J. Nucl. Mater.*, 34, 351 (1970).

3. Effect of Fuel Densification and Relocation on Thermal Performance, *R. A. Proebstle, R. B. Elkins, M. O. Marlowe, E. B. Johamson, G. A. Potts, R. Stijovic, T. C. Rowland (GE-San Jose)*

An irradiation history-dependent model was developed to predict the thermal performance of oxide fuels during irradiation, including the effects of in-reactor fuel densification, fuel swelling, and fuel relocation. The recognition and incorporation of those dynamic effects in the thermal analysis of the fuel have provided favorable comparison of predictions with a wide range of experimental measurements and estimates of fuel temperatures.

The kinetics of in-reactor densification of the fuel are predicted by a model based on the conclusion[1,2] that irradiation-induced diffusion plays a dominant role. Estimates of the rate-controlling irradiation-induced diffusivity in the cool outer region of the fuel, which establishes the effective fuel pellet diameter, were obtained from a comparison of in- and out-of-reactor fuel densification measurements,[2] and have been confirmed[3] by in-reactor diffusivity experiments. The estimate of fuel densification behavior is based on experimentally determinable densification and grain-growth rate constants which are obtained from the response of the fuel to heat treatment. These rate constants are dependent on the fuel microstructure and fabrication history. The predictions of the fuel densification model have been compared with in-reactor fuel column length changes for a wide range of microstructures[4] (e.g., 87 to 95% dense, 2.4- to 22.3-μm grain size sintered at from 1300°C for 1 h to 1600°C for 5 h, with and without pore formers). Good agreement was obtained, generally within 1% of theoretical density to significant burnup.

The swelling of the fuel due to accumulation of fission products was treated as a constant swelling increment per unit exposure.[5] No external (diametral) dimensional changes of the fuel are calculated to occur until the pore volume and any fuel pellet dish volume are filled.

The fuel relocation was formulated as a function of burnup, power, and gap size in a physically reasonable way. Below a threshold power, presumably below which no fuel cracking occurs, no change in fuel cladding gap is predicted. The number of operating cycles was not explicitly considered in the model, since it is difficult to separate effects of power cycles and burnup from available data.

The numerical values of the parameters in the model were found by fitting the model to postirradiation gap-size data. Those data were corrected for estimated effects of fuel densification, cladding creepdown, and irradiation swelling to obtain the gap closure caused by relocation alone.

The thermal conductance of the fuel cladding gap was modeled using the modified Ross-Stoute[6] formulation, with appropriate modification of the conductivity of the filler gas as a result of contamination with fission gas released from the fuel.

The available industry data employed for qualification of the model predictions of fuel cladding thermal conductance included known fuel temperatures as established by (a) direct thermocouple measurement of in-reactor fuel centerline temperatures,[7,8] and (b) postirradiation measurement of the extent of fuel microstructural change[9-12] (equiaxed grain growth, fuel melting). The combined data set represented a heat rating range of 1 to 46 kW/ft, exposure range of ~0 to 10,000 MWd/ST, and initial diametral gap size of 2 to 26 mil. The model predictions were in good agreement with the complete range of available data. Comparison of calculated and observed temperatures showed that the model conservatively predicted 80% of the data with a conservative bias of 120°F for the fuel cladding temperature drop.

1. M. O. MARLOWE, "Predicting In-Reactor Densification Behavior of UO_2," *Trans. Am. Nucl. Soc.*, **17**, 166 (1973).
2. M. O. MARLOWE, "In-Reactor Densification Behavior of UO_2," NEDO-12240, General Electric Co. (1973).
3. A. HOH and Hj. MATZKE, "Fission-Enhanced Self-Diffusion of Uranium UO_2 and UC," *J. Nucl. Mater.*, **48**, 157 (1973).
4. A. HANEVIK, P. ARENSEN, and K. D. KNUDSEN, "In-Reactor Measurements of Fuel Stack Shortening," paper presented at BNES Meeting, London (Oct. 1973).
5. T. C. ROWLAND, M. O. MARLOWE, and R. B. ELKINS, "Fission Product Swelling of BWR Fuels," *Trans. Am. Nucl. Soc.*, **18**, 124 (1974).
6. A. M. ROSS and R. L. STOUTE, "Heat Transfer Coefficients Between UO_2 and Zircaloy-2," CRFD-1075, AECL-1552, Atomic Energy of Canada Ltd. (1962).
7. G. KJAERHEIM and E. ROLSTAD, "In-Pile Determination of UO_2 Thermal Conductivity Density Effects, and Gap Conductance," HPR-80 (1967).
8. "HBWR Quarterly Progress Report, January to March, 1968," HPR-199 (1968).
9. M. F. LYONS et al., "UO_2 Fuel Rod Operation with Gross Central Melting," GEAP-4264, General Electric Co. (1963).
10. T. C. ROWLAND, "UO_2 Irradiation for ORNL, Final Report," GEAP-5642, General Electric Co. (1968).
11. R. N. DUNCAN, "Rabbit Capsule Irradiation of UO_2," CVNA-142 (1962).
12. R. N. DUNCAN, "CVTR Fuel Capsule Irradiations," CVNA-153 (1962).

4450 (CONF-730942-, pp vp PAPER C 1/8) MATHEMATICAL TREATMENT OF THE MECHANICAL DENSIFICATION OF REACTOR FUEL. Rashid, Y. R.; Tang, H. T.; Johansson, E. B. (General Electric Co., San Jose, CA). 1973.

From second international conference on structural mechanics in reactor technology; Berlin, F. R. Germany (10 Sep 1969).

The general phenomenon of mechanical densification of uranium oxide fuel pellets, usually referred to as hot pressing, is characterized by plastic volume change exhibited by the fuel during reactor operations. Much of the fuel behaviour under instantaneous plastic flow and creep, including crack healing and fuel redistribution, can be related analytically to hot pressing. The general problem of the elastic-plastic-creep behavior of light water reactor fuel is examined by first describing a mathematical model for the material's instantaneous and time dependent inelastic response. Then a method for conducting out-of-reactor experiments for evaluating the model's parameters is described. An incremental stress-strain relation which can be used in structural analysis is derived. (auth)

NSA, vol. 29, 1974

56597 (ORO-4212-12) QUANTITATIVE DETERMINATION OF THE STRUCTURE: PROPERTY RELATIONSHIPS IN NUCLEAR FUEL ELEMENT MATERIALS. Quarterly Progress Report, March 16, 1972-June 15, 1972. Rhines, F. N.; DeHoff, R. T.; Whitney, E. D. (Florida Univ., Gainesville. Engineering and Industrial Experiment Station). Aug 1972. Contract AT-(40-1)-4212. 46p. Dep. NTIS.

Creep studies of sintered nickel have advanced to the stage where it is necessary to separate contributions of creep and sintering to overall shrinkage. In order to determine the contribution of sintering to dimensional changes in creep specimens, density-time curves under zero load have been determined. Additional nickel creep data have been successfully modeled by a simple relationship between creep rate and time under load. The reduction of the data to a description in terms of three empirical constants will make tractable correlations between structure, load, and temperature which might otherwise defy reduction from the raw data. Isothermal and isochromal series have been prepared by pressure sintering UO_2 powders from three size fractions. Densification was characterized by dynamic shrinkage measurements at constant heating rate. Microstructural characterization in terms of the metric properties, volume fraction, surface area, and total curvature have been performed on pressure sintered UO_2 specimens. Paths of microstructural change for cold compacted and sintered, and pressure sintered UO_2 specimens were compared. In conventional sintering the path of microstructural change is determined by the compaction pressure. Recent results suggest that temperature may play a role in determining the path of microstructural change for pressure sintered UO_2 structures. Mechanical property measurements have been made on virtually all UO_2 specimens prepared to date. Fracture stress-volume fraction porosity data for specimens prepared by cold pressing and sintering were successfully modeled by a simple exponential relationship involving two empirical constants. Similar curves for pressure sintered UO_2 specimens showed definite maxima with strength dropping rapidly beyond the maximum in each case, presumably due to extensive grain growth. Thermal conductivity data was obtained for a broad range of UO_2 sintered microstructures. The data indicate that thermal conductivity is a function of the initial particle size of volume fractions of porosity up to 0.15. (auth)

NSA, vol. 26, 1972

28545 (ORO-4212-11) QUANTITATIVE DETERMINATION OF THE STRUCTURE-PROPERTY RELATIONSHIPS IN NUCLEAR FUEL ELEMENT MATERIALS. Annual Progress Report, June 15, 1971-March 15, 1972. Rhines, F. N.; DeHoff, R. T.; Whitney, E. D. (Florida Univ., Gainesville. Engineering and Industrial Experiment Station). Contract AT(40-1)-4212. 77p. Dep. NTIS.

The range of available microstructural states in both UO_2 and nickel sinterings has been significantly increased by the fabrication and acquisition of hot pressing equipment, as well as by progress in separating fractions of the as-received powder. Results of microstructural characterization in terms of averages of the metric properties, volume fraction, surface area, and total curvature indicate that the microstructural states thus far obtained generally fall nominally within the range of those observed for other equiaxed powders. UO_2 structures prepared by cold pressing deviate from this sintering behavior because the effects of cold pressing were not removed by subsequent treatments. Property measurements made on characterized microstructures have yielded empirical quantitative correlations for gas permeability and tensile strength of porous nickel. Relations between properties of fracture surfaces and mechanical behavior have also been evolved. The determination of variation of thermal conductivity with structure for one series of samples has demonstrated that modifications of the thermal conductivity apparatus to allow atmosphere control have been successful. The analysis of creep in nickel sinterings has been greatly facilitated by the observation of simple relationships between creep rate and time under load. The reduction of the data to a description in terms of three empirical constants will make tractable correlations between structure, load, and temperature which would be too complex to deduce from the raw data. (auth)

NSA, vol. 26, 1972

13098 BABCOCK AND WILCOX'S IRRADIATION PROGRAM ON FUEL DENSIFICATION. Rigdon, M. A.; Papazoglou, T. P.; Baroch, C. J.; Montgomery, M. H. (Babcock and Wilcox, Lynchburg, VA). pp 59.1-59.4 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004-.

Babcock & Wilcox is conducting a two-phase research program to investigate the fuel densification phenomenon. The effects of accumulated burnup, compressive stress, and fuel temperature on the densification of commercially produced UO_2 fuel pellets operating at linear heat rates from 3 to 5 kW/ft (98 to 165 W/cm) are being investigated in Phase 1. The effects of accumulated burnup, linear heat rate, and pore size distribution on the densification of specially manufactured UO_2 pellets with carefully controlled microstructures will be investigated during Phase 2. Four commercial fuels have been irradiated up to 1600 hours at linear heat rates ranging from 3.5 to 4.5 kW/ft (115 to 150 W/cm). The surface temperatures of these fuels are estimated to range from 700 to 800 F (370 to 420 C). Measurements taken from neutron radiographs indicate that no fuel columns have experienced shortening under these conditions. (5 references) (auth)

NSA, vol. 32, 1975

31759 (WCAP-8388) PERFORMANCE CHARACTERISTICS OF MIXED PuO_2-UO_2 FUELS IN PRESSURIZED WATER REACTORS. Rim, C. S.; Chubb, W.; Caye, T. E.; Argall, B. M. (Westinghouse Electric Corp., Pittsburgh, Pa. (USA). Nuclear Energy Systems Div.). Feb 1975. vp. Westinghouse Electric Corp., Pittsburgh, PA.

Usage of mixed PuO_2-UO_2 fuel in pressurized water reactors is expected to increase significantly in the near term. Physical properties and performance characteristics of mixed oxide fuel are presented. Comparisons with uranium dioxide fuel are included. Westinghouse plutonium utilization experience in operating pressurized water reactors (PWRs) is also summarized. Evaluation of nuclear and thermal-hydraulic core characteristics, and reactor safety analysis for core designs employing mixed oxide fuel assemblies are covered appropriately on a case by case basis. (49 references) (auth)

NSA, vol. 31, 1975

Controlled Porosity Oxide Fuel, *H. Rogan, T. J. Heal, J. E. Littlechild, L. Raven (British Nucl Fuels Ltd–UK)*

THE CONCEPT OF CONTROLLED POROSITY FUEL

Controlled porosity is a term given to the tailoring of oxide fuel microstructures to fit them for a particular purpose. The aim of the microstructure control is to provide a fuel pellet which is:

1. densification resistant
2. swelling resistant
3. exhibits the lowest possible gas release
4. has the lowest possible gas and moisture content
5. is readily and reproducibly made to tight specifications in pellets of any size or density.

A review of pore stability both in and out of reactor leads to the conclusion that densification resistance will be exhibited by fuels containing the minimum possible amount of porosity as pores $<\sim 5$ μm (Ref. 1). However, most sintered UO_2 pellets commonly contain substantial porosity as pores <5 μm. BNFL have therefore developed a technology which gives a structure of closed large pores approximately randomly distributed in a high density matrix of UO_2 containing $<2\%$ of ~ 5 μm porosity also as closed pores. The essential features of the technology are the use of particulate pore formers and of a UO_2 powder which sinters quickly to a very high density. The overall pellet density can be any required value.

As such a structure contains a minimum of small porosity, it will be densification resistant in-reactor, the large pores will persist and will still be present in the structure at the onset of swelling, except in the columnar grain growth region where the large pores will be swept to the pellet center in the usual way.

When swelling becomes significant, the nucleation of gas bubbles on grain faces and edges and the solid fission product swelling within grains will, combined with any fuel/cladding interaction, place the large voids under substantially hydrostatic compression. Elementary treatments of the large porosity by the normal surface tension approach shows that the large pores are unstable to pressure and the swelling UO_2 will creep into the pores at a rate governed by the stress and the UO_2 creep rate, which will in turn be dependent on fuel temperature, rating, fission product content, and grain size, as well as pore size and spacing. In this way, the controlled porosity can accommodate fission product swelling, although the precise extent of the accommodation cannot yet be predicted from theory.

As the pellet when loaded into the reactor contains essentially no open porosity, the surface available for gas release is minimized; whereas this advantage is, of course, immediately lost in the columnar grain growth region, and ultimately lost above ~1000°C as fission gas bubbles interlink, the cooler annulus will retain a significantly longer mean-free-path to a free surface than fuels with open porosity and the effect of this in lowering the release from the pellet as a whole may be significant in certain circumstances.

The lack of open porosity in the pellet ensures that the surface area available for moisture and gas adsorption is small and calculations on the amount of gas that can be present in pores indicate a hydrogen content of only ~0.2 ppm, even if all the pores are filled to the maximum possible pressure with pure steam.

FUEL MANUFACTURE

The first requisite for the operation of a controlled porosity route is an oxide which will sinter to a high density in the shortest possible sintering time. The BNFL dry process for conversion of UF_6 to UO_2—the integrated dry route (IDR)—can readily provide such an oxide. In this process, UF_6 is reacted with steam in the gas phase at the feed end of a rotary kiln to give an active UO_2F_2 which is converted in its passage down the kiln to ceramic UO_2 in a countercurrent flow of H_2 and steam.

The powder, which is not free flowing as produced, is made up of small (1 to 2 μm) aggregates of crystallites of more or less spherical shape and a diameter of 0.15 to 0.4 μm. Surface areas calculated from the observed crystallite size distribution correlate well with BET surface area measurements which generally lie between 1 and 3.5 m^2/g.

BNFL at present uses a binder pelleting route and for this purpose the powder is normally micronized to render it more suitable for granulation. The milling process merely breaks down the aggregates leaving up to 90% of the powder below 1 μm in diameter.

When the powder is to be used for binderless fabrication, milling is not required, and it has been found that control of the crystallite shape is possible, a more dendrite-like shape being produced which produces much stronger green compacts than the spherical crystallites.

Optimization of the process to produce oxides of the highest possible sinterability yields powders that attain densities in the region of 98% theoretical density when sintered for 3 h at 1650°C and densities up to 99% theoretical density can be produced under commercially acceptable sintering conditions.

Having optimized a powder, a controlled porosity additive must be found which is capable of reproducibly introducing voids of the required size and amount with no deleterious effect in the sintering of the oxide and on the pellet microstructure, and which should be usable without modifying pelleting processes.

Extensive development in BNFL has led to the establishment of a family of additives, known by the generic name of CONPOR, which are each designed to be compatible with an existing pellet route. Thus, though BNFL at present use a binder route for large-scale pelleting, and have additives which are compatible with this route, other members of the CONPOR family can be used in conventional binderless fabrication with no process modification whatsoever, and this has enabled other manufacturers to process BNFL oxide with controlled porosity through their pelleting lines with excellent results.

(Cont'd pg 108)

FUEL QUALITY

To define a controlled porosity fuel, several parameters must be measured. These include matrix density, pore size distribution, porosity homogeneity, and thermal stability; the measurement of these parameters will be discussed in the presentation of the paper. It may be stated that all these parameters can be controlled and typical fuels have matrix densities in excess of 98.5 theoretical density, porosity homogeneity better than random, median pore size controllable within narrow limits in the range 10 to 300 μm, and a density stability of better than 1% change after 24 h at 1700°C. Moisture and gas contents are low, with total hydrogen contents in the range 0.1 to 0.2 ppm. A typical production BWR pellet is shown in Fig. 1.

IRRADIATION PERFORMANCE

It is important to remember that a primary reason for selecting a fuel with less than maximum density is to provide voidage to accommodate swelling due to fission products formed during the irradiation. Current fuel performance models, based on experimental evidence, enable us to predict the amount of voidage required for a particular duty and the fuel is manufactured so as to provide the required amount of voidage. It is implicit in this argument that the voidage is there when it is required; it must not disappear early in life by any in-reactor densification process. To test the effectiveness of CONPOR fuel, we have mounted a series of irradiation experiments, the first being loaded into SGHWR at Winfrith in 1967. Others have been irradiated in various reactors; some have subsequently been examined. There are several under irradiation at the present time and further experiments are planned. All this work is aimed at understanding the process of densification and swelling, and optimizing the pore size, shape, and distribution in relation to the UO_2 grain size for a given amount of initial voidage.

POSTIRRADIATION EXAMINATION

Table I gives some results obtained from two experimental fuel assemblies. They show little change in pellet density for either low rated fuel with no gas bubble swelling, or for high rated fuel containing a significant amount of grain boundary gas bubbles.

The whole pellet density measurements were made on samples of irradiated fuel using the mercury pyknometric technique which was also used on the unirradiated fuel. Twenty-gram samples were used, with two determinations per sample. (See Table I.)

TABLE I
Density Measurements on Whole Pellets of Controlled Porosity UO_2

Assembly	Rating kW/ft	Time Equivalent Full Power Days	Burnup MWd/MTU	Preirradiation Density, g/cm^3 (mean)	Postirradiation Density, g/cm^3	Observed Density Increase g/cm^3
S21/1	5.6	116	1200	10.09 ± 0.01	10.13 10.13	0.04
	8.0	116	1670	10.09 ± 0.01	10.10 10.11	0.015
	9.0	116	1900	10.09 ± 0.01	10.13 10.15	0.05
	12.0	116	2500	10.09 ± 0.01	10.13 10.13	0.04
J03/1	5.6	186	1900	10.08 ± 0.04	10.14 10.14	0.06
	8.0	186	2700	10.08 ± 0.04	10.08 10.06	-0.01
	11.0	186	3600	10.08 ± 0.04	10.11 10.10	0.025
	12.0	186	4100	10.08 ± 0.04	10.13 10.13	0.05
	14.0	186	4800	10.08 ± 0.04	10.14 10.16	0.07
	14.0	186	4900	10.08 ± 0.04	10.05 10.06	-0.025

Transverse sections were taken adjacent to the density specimens and photomicrographs prepared from the sections. Comparison of the sections showed that:

1. Densification of small pores was the same for fuel operating at the same temperature but different ratings.

2. It was not the same for fuel at the same rating but different temperatures.

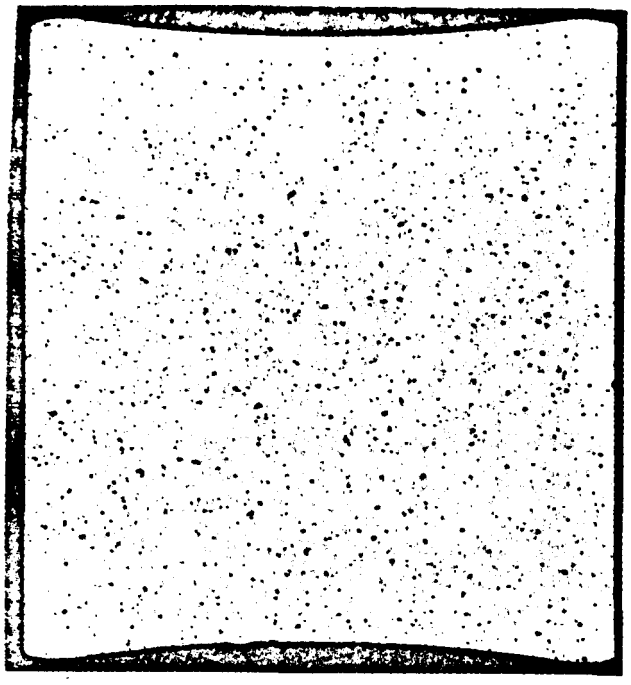
3. Above about 1100°C, all original microporosity was lost and the grains appear to be 100% dense.

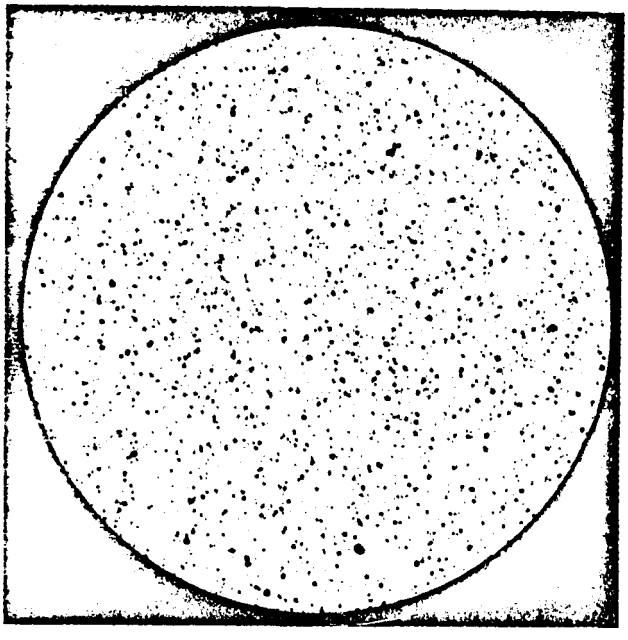
4. The larger (>5 μm) induced pores persist at all temperatures up to about 1700°C. They are, however, progressively reduced in size under the forces imposed by fission product swelling. Above 1700°C, the columnar grain growth sweeps them toward the center to eventually form a central void.

The density chosen in the case of the two experimental fuel assemblies cited above 10.08 g/cm^3 was selected to show the advantage of the controlled porosity technology in resisting densification and absorbing fission product swelling.

Computer programs have been developed to predict the changes described above and to take account of the restraint imposed by internal gas pressure or the fuel cladding.

1. T. J. HEAL, J. E. LITTLECHILD, and R. H. WATSON, "Development of Stable Density UO_2 Fuel," BNES Symp. Nuclear Fuel Performance, Paper 52 (Oct. 1973).





Sintered density - 10.30 g/cm^3
Density after further 24 h at 1700°C - 10.39 g/cm^3

Fig. 1. Typical controlled porosity BWR pellet.

1. Model for Prediction of Fuel Failures, *E. Rolstad (Scandpower-Norway)*

The fuel performance models used by the vendors in design, marketing, and licensing have made no attempt so far at predicting the number of fuel failures that will occur in the reactors.

This situation has been understandable because the failures that occurred were not expected and were therefore unpredictable based on earlier knowledge. Examples of this are the early hydride failures and, later, the ones that resulted from fuel densification. These two special problems have now been solved, but the third problem, the PCMI failures, still exists and probably will for many years. Experience with PCMI failures is now quite extensive, however, and systematic trends have emerged. The failures are happening as a result of large and rapid power increases (power shocks) produced by increases in the total or local power to a value beyond that at which the fuel usually operates, and the problem increases with burnup.

Even if it is correct to say that the failures are systematic, a stochastic element seems to play an important role in failure occurrence. Failures can therefore occur in power reactors at power shocks smaller than half the shock threshold determined by experiment in test reactors. The failures probably occur when a number of important parameters happen to be at the most unfavorable side of their tolerance bands, or when the most unfavorable fuel-crack pattern happens to coincide with the worst cladding defect. Models like CYGRO or the FRAP code, for example, have to work with nominal or average values of dimensions and properties to predict the most likely thermal and dimensional behavior of the fuel rods. These, therefore, are not suitable for prediction of power shock failures.

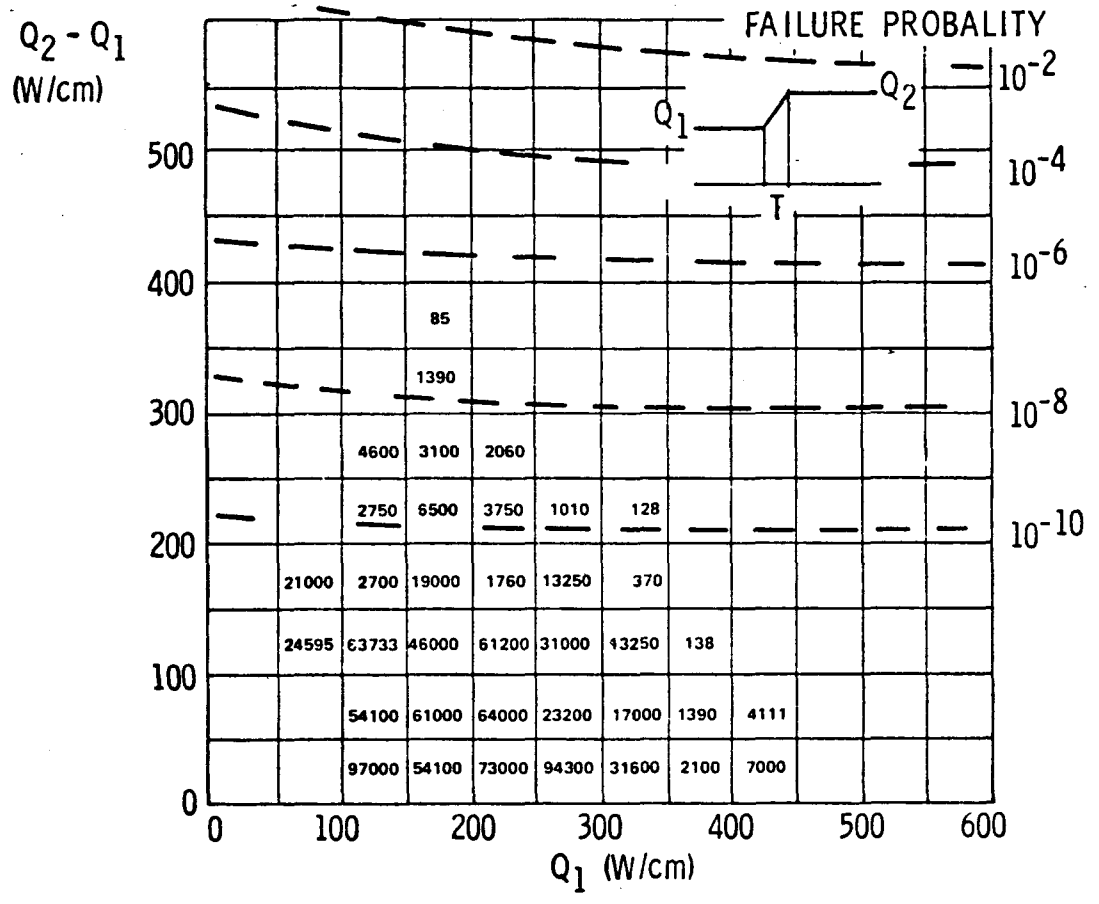


Fig. 1. Illustration of a power shock matrix with failure probability curves superimposed.

A model called POSHO, with its only objective being to predict this type of fuel failure, has therefore been developed. The basic assumption in the model is that thermal expansion of the UO_2 during power increases results in tensile stresses in the cladding. The thermal expansion is transmitted as a stress in the cladding only from the power value (Q_o) where the gap between fuel and cladding becomes zero. An important part of the POSHO model is thus to calculate the local Q_o values as the operation progresses. This is done by means of two algorithms; one for the situation when the power $Q < Q_o$ and another for the situation $Q > Q_o$. The first algorithm computes a negative value of dQ_o/dt with a magnitude mainly dependent on Q. The physical meaning of this is an outward relocation of the fuel. The other algorithm computes a positive value of dQ_o/dt whose magnitude depends on fuel design parameters and the value of $Q - Q_o$.

Analysis of a reactor cycle can be done either in the predictive mode based on a planned cycle or on a previous cycle based on a real power history using measured power history, control-rod history, and power distributions.

When a reactor cycle is being analyzed, events during the cycle that result in power shocks are first identified. Startup of the cycle is always an important event as are the sequence operations and sequence extension operations in the BWRs. The local power shocks resulting from each of these events are then calculated and processed through a fuel failure probability routine giving the number of cladding cracks produced and the number of fuel rods and assemblies that have failed during each of the events.

This procedure is illustrated in Fig. 1, where the numbers in each of the squares give the number of pellet interfaces that have been subjected during an event to a power shock by an amount given by the matrix coordinates of the square. The dotted curves give failure probability values. The number of cladding cracks produced is obtained by multiplying the number of pellet interfaces with the corresponding failure probability number.

Figure 1 can represent a complete reactor core, one fuel assembly, or one axial and radial node of an assembly dependent on how detailed the analysis is being done. The probability lines are computed based on fuel design parameters, burnup, and previous power shock failure experience.

The program is in the development stage, but has been "calibrated" against a number of reactor cycles, and shows good qualitative agreement when applied to other cycles not used in the calibration. For two equal BWR reactors, one operated with 7×7 fuel and the other with 8×8 fuel, the program predicts approximately ten times more failures in the 7×7 fueled reactor.

ANS TRANS., vol. 24, 1976

28552 NONSTOICHIOMETRY OF NUCLEAR FUEL MATERIAL. STRUCTURE AND THERMODYNAMICS. Saito, Yasutoshi (Akita Univ., Japan). Nippon Kinzoku Gakkai Kaiho; 10: No. 10-11, 641-52; 742-50(Oct 1971). (In Japanese).

An outline of the nonstoichiometric phase relationship of ceramic fuel material is presented along with an explanation of structure and thermodynamics of uranium dioxide. The ceramic fuel material comprises the uranium dioxide $UO_{2\pm x}$, plutonium dioxide PuO_{2-x}, mixed oxide of uranium and plutonium $(U,Pu)O_{2\pm x}$, carbide of uranium and plutonium, and oxide and carbide of thorium. (Japan)

NSA, vol. 26, 1972

38876 KINETICS OF PORE MOVEMENT IN UO_2 FUEL RODS. Sens, P. F. (Reactor Centrum Nederland, Petten). J. Nucl. Mater.; 43: No. 3, 293-307(Jun 1972).

The movement of pores in UO_2 fuel elements occurs by diffusion of UO_2 vapor molecules across the pores with the temperature and concentration gradients as driving forces. The corresponding diffusion equation was derived and solved numerically throughout the pore volume with the PROFORMA-2 computer program. The pore velocity was found to depend on the shape of the pore and the initial pore shape changed during migration. Disc shaped pores had the highest velocity and transform, in moving, to the commonly observed lenticular voids, pinching off small voids at its circumference. The net velocity of these pores as a function of temperature and temperature gradient is given and applied in the POREMOVSKY computer program to calculate the redistribution of porosity, power generation and temperature at different time intervals. Generally, good agreement was found between calculated and observed microstructural changes, the majority of which appeared to take place early in the irradiation period. (auth)

NSA, vol. 26, 1972

1. Impact of Fuel Densification on the U.S. LWR Industry, *L. S. Rubenstein, D. F. Ross (USAEC)*

Fuel densification came into notoriety in the Spring of 1972, when cladding collapse was observed in the fuel of a PWR. As a result, the following happened, in order: (a) an AEC report on Fuel Densification and its Effects[1]; (b) analyses of the effects of fuel densification on light-water reactors, which in some cases involved reductions in permissible operating powers and other plant restrictions; (c) reanalyses by fuel manufacturers based on a wider data base than in Ref. 1; and finally, (d) improvements in the fuel design. An evolution from problem to interim solution to near-final solution has taken place in the past two years.

The effects of fuel densification[1] are to:

1. shrink the pellet radially, causing an increase in the pellet-cladding gap, a decrease in the gap conductance, and a consequent increase in the pellet average temperature (the stored energy) at fixed power production
2. shrink the pellet axially, causing an increase in the linear heat generation rate
3. create the possibility of a gap in the fuel column from pellet hangup, following pellet axial shrinkage, with attendant increase in the local fission rate in "sound" pins near a gap (the power spike)
4. create the possibility of cladding creep collapse into a gap in the pellet column.

The limits on operating conditions were sufficient motive for the LWR industry to modify their fuel designs and to improve analytical techniques. Some of the proposed improvements in fuel design are:

1. stable (i.e., little-densifying) fuels
2. physical and empirical models to quantify the rate and extent of densification
3. redesign of fuel pellets to minimize fuel cladding interaction and, thus, pellet hangup and gap formation
4. increased fill-gas pressure, for some PWRs
5. improved pellet sintering conditions and increased initial density
6. improved analytical models for integrating densification and other phenomena into a predictive technique for stored-energy calculations.

Application of some of these improvements has already been accomplished, with corresponding removal of some of the operating limits. It is not expected that densification will "go away" entirely. Rather, it is expected that, in the near future, sufficient data will be on hand to remove the conservatisms imposed in Ref. 1.

1. "Densification of Light Water Reactor Fuels," Regulatory staff, USAEC (Nov. 14, 1972).
2. "AEC Interim Acceptance Criteria for ECCS" (June 1971).

These effects were analyzed by fuel manufacturers and reactor owners in the late 1972-early 1973 time span. In some cases, the operating plants had to reduce their core thermal power. The reduction was based on the following scenario:

1. Calculate instantaneous fuel densification to a theoretical density of 96.5% geometric,[1] with proportional decrease in radius and increase in stored energy.
2. Calculate the loss-of-coolant accident (LOCA) consequences, with the end product being local or regional linear heat rate sufficient to meet the AEC interim acceptance criteria (IAC).[2]
3. Calculate the permissible plant operating conditions, taking into account the axial shrinkage and the power spike.
4. If cladding creep collapse is calculated, do steps 1 through 3, but for a maximum cladding temperature of 1800 rather than 2300°F.

In some of the limiting cases, it was seen that cladding creepdown onto the pellet would affect the stored energy calculation and thus the peak cladding temperature limited by the AEC IAC to 2300°F to account for the design-basis LOCA. Hence, the permissible linear heat rate would increase with time. In recognition of this, the Regulatory staff would issue Technical Specifications on permissible linear heat rate that varied with exposure. Some of these restrictions were still in effect as of February 1974.

ANS TRANS, vol. 18, 1974

4. In-Reactor Restructuring Studies of UO_2, *Jean-Pierre Stora, Claude Hueber (CEN-Grenoble)*

The Furet program, undertaken in 1970 with the view of studying the recrystallization of UO_2, has led to the establishment of a complete restructuring model of the fuel. Twenty-four Zircaloy capsules have been irradiated between 1970 and 1973 over very short time scales (6 h to 14 days) in the Siloe pool reactor at Grenoble.

The capsule is a thick tube of Zircaloy-2, 13-mm i.d., 7 mm thick, containing UO_2 pellets of density 10.57 g cm^{-3} and 12.9-, 12.7-, 12.5-mm diam. The length of the fuel stack is 150 mm. Half of the capsules have a central W5%Re/W26%Re thermocouple in the oxide.

As a consequence of the precise measurements of thermal power and temperatures, the integrated thermal conductivity of UO_2 has been measured between 50 and 1400°C. The previous results of conductivity in the Cyrano program between 300 and 2300°C (Ref. 1) are in agreement to better than 1% with the Furet measurements over the common range of temperature.

The structure modification of UO_2 of 9.5 density without diametral clearance in operation is due to pore movement. The measurement of the displacement of the lenticular pores, the origin of which is the initial porosity of the sintered UO_2, verifies the Nichol's expression which predicts that the radial velocity passes through a maximum.[2]

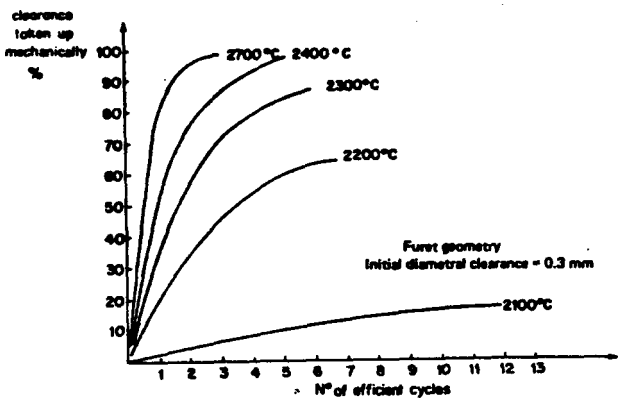


Fig. 1. Number of efficient cycles to mechanically take up the fuel-to-sheath clearance.

The irradiations with diametral clearance are representative of the utilization of UO_2 in water reactors and have led to the establishment of a restructuring model based on the following steps:

1. The geometry at temperature and the cumulative width of the radial fissures, e(r), at radius, r, are calculated. The restructuring by evaporation-condensation processes leads to some displacements of mass which give a central void, an outer zone which remains fissured, an inner zone restructured and without cracks at temperature. The mechanisms involved in the new structure are:

   (a) annealing of circumferential cracks

   (b) filling in radial cracks

   (c) migration of porosity.

The rate of healing of a circumferential crack and the concomitant dedensified zone are calculated. The filling in of radial cracks by evaporation-condensation processes and the porosity migration by diffusion are taken into account to explain the formation of the central void and to calculate its radius.

2. A consequence of the movement of matter due to the evaporation-condensation processes is that the thermal contraction of the restructured fuel cannot be reversible: the diametral clearance is taken up. Healing of a circumferential crack takes up clearance equaling the width of the crack, while filling in a radial crack of width, f_r, takes up $\delta_r = f_r/2\pi$. Experiments show that thermal cycling, which plays an important role in the take-up of the diametral clearance, is efficient only if the cracks are annealed at convenient temperatures and times for restructuring. This introduces the notion of consolidation of the oxide fragments. For example, Fig. 1 gives the number of efficient thermal cycles for mechanically taking up the diametral clearance in percent.

The restructuring model allows calculating the structural state of the fuel, the take-up of diametral clearance (therefore the operating clearance), and the fuel-to-cladding heat transfer coefficient at each step of irradiation. It is then possible to calculate the temperature distribution across the fuel at any given moment; therefore, the fission gas release and the resulting heat transfer modifications. It is also possible to predict and to calculate the mechanical fuel-to-can interactions. Modeling these results is introduced into a computer program.

1. J. P. STORA and P. CHENEBAULT, "Mesure de l'Intégrale de Conductibilité Thermique d'UO_2 Fritté Jusqu'à 2300°C. Evolution des Gaz de Fission à Puissance Constante," CEA.R.3618 (1968).

2. F. A. NICHOLS, "Pore Migration in Ceramic Fuel Elements," *JNM*, 27, 2 (1968).

16530 URANIUM DIOXIDE PROPERTIES FOR LWR FUEL RODS. Stehle, H.; Assmann, H.; Wunderlich, F. (Kraftwerk Union A. G., Erlangen (F. R. Germany)). Nucl. Eng. Des.; 33: No. 2, 230-260(Sep 1975).

127 refs.

Uranium dioxide pellets have become the most important nuclear fuel, and will remain so for a long time, with the fissile isotope ^{235}U being replaced by PuO_2 additions. This does not significantly change the pellet properties. Uranium dioxide properties affect fuel rod performance more than previously anticipated, because UO_2 pellets show a distinct response to irradiation, and because of mechanical and chemical interaction with cladding. Here elastic and plastic behaviour, fracturing, irradiation, densification, and dimensional behaviour under steady and power cycling conditions are mainly covered. (NL)

NSA, vol. 33, 1976

23865 (BAW-10055) FUEL DENSIFICATION REPORT. Turner, R. A. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Nuclear Power Generation Div.). Feb 1973. 71p.

The Regulatory Staff of the U. S. Atomic Energy Commission has concluded that adequate data are available to account for the effects of fuel densification in the safety evaluation of nuclear power reactors. Considerable reliance was placed on the direct application of data obtained from operating reactors. These reactors have been operated safely with densified fuel and collapsed cladding, and the Regulatory Staff has concluded that implementing the recommendations of their report, "Technical Report on Densification of Light Water Reactor Fuels," will provide increased assurance of safe operation of reactors in which fuel densification is expected to occur. The AEC's recommendations and guidelines have been implemented and are described. (auth)

NSA, vol. 27, 1973

26596 (BAW-10055(Rev.1)) FUEL DENSIFICATION REPORT. Turner, R. A. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Nuclear Power Generation Div.). Jun 1973. 106p. Dep. NTIS $7.50.

The Regulatory Staff of the U. S. Atomic Energy Commission has concluded that adequate data are available to account for the effects of fuel densification in the safety evaluations of nuclear power reactors. Considerable reliance was placed on direct application of data obtained from operating reactors. These reactors have been operated safely with densified fuel and collapsed cladding, and the Regulatory Staff has concluded that implementation of the recommendations of their report, "Technical Report on Densification of Light Water Reactor Fuels," will provide increased assurance of safe operation of reactors in which fuel densification is expected to occur. The AEC recommendations and guidelines have been implemented and reported herein. This report has been revised to incorporate the revised power spike model based on results obtained from Brookhaven National Laboratory. In addition, several minor corrections have been made for clarity and accuracy. Appendix A contains the answers to AEC questions which resulted from the review of B&W fuel and are included in this report for clarity and completeness. (auth)

NSA, vol. 28, 1973

PWR Fuel Operating Experience and Performance Evaluation Program, *R. Traccucci (Framatome–France)*

INTRODUCTION

The fuel assembly design is justified by four principal means: analytical studies, out-of-pile testing (mechanical and hydraulic testing), in-pile testing in test reactor, and finally operating experience in commercial reactors. Although the three first means are necessary, it is sure that the last means is unique to confirm completely and at the same time the design and the performance of the fuel element. As a matter of fact, performance of the fuel element, and particularly the fuel rod, depend on the design and the quality control executed during manufacturing, but also on the plant operating conditions (water chemistry, power history, control rod motion). Although Framatome action covers all these means, we will be limited in the presentation to the last one; i.e., the effort by Framatome to evaluate the in-pile fuel element performances. It should be pointed out that such efforts impose close collaboration between the fuel vendor and the utility, as well to define the objectives to realize the irradiation and on-site examination programs. Besides, Framatome, as a Westinghouse licensee, profits from the accumulated experience. Therefore, the first part of this presentation deals with Westinghouse fuel operating experience; the second part deals with the description of the fuel performance evaluation program.

OPERATING EXPERIENCE

The operating experience accumulated up to now has shown good performance, in general. However, a small percentage of fuel elements has been defective. In any case, the defects have had no consequence on the plant safety, the activity level being less than the technical specification limits.

Three observed conditions causing defects leading to cladding fissures are: high impurity levels (moisture) in the pellets leading to cladding internal hydriding, fretting due to high local velocity jets coming from the core baffle, and finally, high pellet densification and the corresponding cladding collapse. Another anomaly encountered is axial rod bowing due to irradiation growth and insufficient gap between fuel rods and nozzles. In addition, bowing can be caused by other means. Successive design improvements have suppressed these defects and only a small percentage of fissured rods have been detected in reactors recently put into operation. Their frequency is very small.

(Cont'd pg 115)

Description of the main observed defects and their respective remedies are presented:

Internal Hydriding

Cladding internal hydriding leading to cladding fissures has been observed in different reactors. The defects appear after about 1000 EFPH. Their number increases quickly during a one-month period and then stabilizes. This defect has been seen in the first fuel element generation using low density pellets. Density increase and the use of more restrictive pellet moisture specification before loading in fuel rods have resolved the problem.

Fretting Corrosion on Peripheral Fuel Assemblies

High velocity coolant jets through gaps in the core baffle have led to vibrations and, then, rod ruptures. After repair of the core internals, this type of defect has disappeared.

Densification

Important fuel densification has appeared in the first generation of Zircaloy-clad elements, characterized by low pellet density. Densification leads to fuel column shrinkage and the statistical formation of voids. At void levels, the unsupported cladding first creeps down and finally collapses. In most cases, in particular in the case of pressurized rods, no collapse has been seen. However, in-core instrumentation has located local flux peaking in fuel rods adjacent to rods presenting column voids (due to reduction in neutron absorption).

Fuel column shrinkage leads to an increase in the average linear power and the local peak leads to an increase in the maximum linear power, which, combined with the pellet temperature increase, can induce core power capability constraints.

The pellet density increase and a best choice of manufacturing processes leading to pellet microstructure optimization minimize in-pile densification. Pressurizing of fuel rods has greatly reduced the probability of cladding collapse. The residual densification is taken into account in the analytical model used for the core design and safety studies.

Fuel Rod Bowing due to Interference

This defect has also appeared in the first generation of Zircaloy-clad fuel. The clearance between rod ends and the top nozzle was not sufficient to compensate for the axial rod growth. Rod and top nozzle interference has been observed during visual examination of the discharged fuel. The clearance increase has solved the problem.

Noninterference-Related Rod Bowing

Visual examination on discharged fuel has shown local fuel rod deformations between grids. The frequency of occurrences is very small, particularly for large deformation amplitude. These deformations are mainly related to the axial compression stresses induced by the grid spring friction on cladding during its irradiation growth and to subsequent cladding creep. Nevertheless, a certain number of other mechanisms can be related to these deformations. They are:

1. the flux and temperature gradient across the fuel assembly leading to an azimuthal temperature variation in the cladding
2. the effects of an azimuthal variation of the cladding thickness leading to an azimuthal variation of the axial stresses and strains
3. the effects of an azimuthal variation of the pellet-cladding mechanical interaction, if any.

The observation of fuel assemblies in different reactors supports the grid/rod interaction mechanism. Particularly, it has been verified that the deformation frequency and amplitude have been reduced by the presence of a clearance between the bottom rod end and the bottom nozzle, and by the decrease of grid spring forces.

The present standard assembly design with axial clearance at both fuel rod ends then minimizes this phenomenon. Nevertheless, core thermal-hydraulic analysis has been performed to account for this situation.

PERFORMANCE EVALUATION PROGRAM

To verify fuel element predicted performances, in particular, those directly connected to densification and rod bowing phenomena, on one hand, and to determine the causes of other defects presently not explained and eventually improve the fuel element design and the plant operation technical specifications, on the other hand, Framatome is undertaking, in cooperation with the utilities, particularly EdF, a performance evaluation program. This program comprises essentially post-irradiation on-site nondestructive examinations, special fuel assembly irradiation, hot cell destructive examination, analysis of operating conditions, and, finally, analysis of examination results and the eventual derivation of new analytical models.

A more detailed description of the means necessary to accomplish this program is given:

1. examination in the spent fuel pool of standard assemblies with use of specialized equipment, like an endoscope connected to a television camera; this examination permits observing the eventual fuel assembly and rod bowing.

2. measuring the fuel column length with in-core instrumentation; this permits following the densification evolution and comparing it to predicted values

3. removable fuel rod assembly irradiation. This assembly is used by Westinghouse in a large number of reactors. It is different from the standard assembly by a modified top nozzle which allows extraction of a certain number of rods. These rods, well precharacterized, can be extracted during refueling periods with a special handling tool, be examined in the spent fuel pool with special examination devices (television, gamma scanning, profilometry, and crud examination), and be reinserted in the assembly or sent to the hot cell for destructive examinations and be replaced by new rods. One removable fuel rod assembly is irradiated in the TIHANGE core. One or two others will be irradiated in one of the first EdF PWR reactors.

4. surveillance of primary coolant chemistry characteristics to derive a correlation between these characteristics and the crud deposit on fuel rods and eventual cladding corrosion

5. the centralized analysis of operating conditions (primary coolant activity and chemistry, power history, control rod motions) to determine with precision the influence of these parameters on fuel element performance.

25964 (WASH-1330) TECHNICAL REPORT ON DENSIFICATION OF WESTINGHOUSE PWR FUEL. (USAEC Regulatory Staff, Washington, D. C.). 14 May 1974. 26p. Dep. NTIS $4.50.

The Westinghouse Fuel Densification and Power Spike Models as described in WCAP-8218 have been reviewed by the Staff. These models were found to be acceptable for the evaluation of densification effects in pressurized PWR fuels that have been manufactured within the processing bounds as described in WCAP-8218. Where the Westinghouse Densification Model predicts a fuel geometric density in excess of 96.5% theoretical density, the Staff approves the option to use the lower density value, i.e., 96.5% TD with the predicted densification rate. (auth)

NSA, vol. 30, 1974

40410 LATTICES OF PLUTONIUM-ENRICHED RODS IN LIGHT WATER. I. EXPERIMENTAL RESULTS. Uotinen, V. O.; Lauby, J. H.; Schmid, L. C.; Stinson, W. P. (Battelle-Northwest Labs., Richland, Wash.). Nucl. Technol.; 15: No. 4, 257-71 (Aug 1972).

Results of experiments conducted in the Critical Approach Facility are presented for a total of 48 lattices of plutonium-enriched rods. Five kinds of UO_2-PuO_2 rods ranging in PuO_2 enrichment from 1.5 wt% to 4.0 wt%, and three kinds of Al–Pu rods ranging in Pu enrichment from 1.8 wt% to 5.0 wt%, were used in the experiments. The lattices covered a broad range of water-to-rod volume ratios. Parameters reported for each lattice are: the number of rods required for a critical loading, the relaxation length measured in an exponential loading, the critical buckling and reflector savings. For five of the lattices results are also reported of the worth of central absorber rods of boron and hafnium. The results of approach-to-critical experiments were verified by comparing them with the results of critical measurements conducted in three lattices. (34 references) (auth)

NSA, vol. 26, 1972

10378 (CONF-740947-4) THERMAL DENSIFICATION OF MIXED-OXIDE FUEL. Voglewede, J. C. (Argonne National Lab., Ill. (USA)). [nd]. 14p. Dep. NTIS $4.00.

From American Ceramic Society meeting; Williamsburg, Virginia, USA (29 Sep 1974).

The out-of-pile stress-assisted densification of mixed-oxide fuel materials is examined. A constitutive equation for final-stage densification of mixed-oxide fuels is presented which is applicable to a wide range of temperature, stress, and density. (JWR)

NSA, vol. 31, 1975

58124 (NP-19451) NON-PROPRIETARY INFORMATION ON NUCLEAR FUEL PERFORMANCE. (Westinghouse Electric Corp., Pittsburgh, Pa.). 18 Sep 1972. 30p.

Tables and curves of performance characteristics of Westinghouse reactor fuels are presented. Proprietary information has been deleted from many of them. (J.M.J.)

NSA, vol. 26, 1972

Radiation Damage in Uranium Dioxide

By A. D. Whapham and B. E. Sheldon
Atomic Energy Research Establishment, Harwell, Berkshire

[Received 15 April 1965, and in revised form 29 July 1965]

Abstract

Point defect clusters and dislocation loops produced in bulk uranium dioxide by fission damage have been observed and studied by transmission electron microscopy. Clusters 25 Å in diameter are first observed at a dose of $4{\cdot}3 \times 10^{15}$ fissions cm^{-3}. With increasing dose, these clusters quickly grow into resolvable prismatic dislocation loops which have been shown to be formed from interstitial atoms. They lie on {110} planes and have a $\frac{1}{2}\langle 110\rangle$ Burgers vector. Evidence of the evaporation of small loops traversed by fission fragment tracks is given, and used to explain the rapid growth of the loops. At high doses the loops grow and coalesce to form a network of dislocations, but at a dose of $1{\cdot}6 \times 10^{20}$ fissions cm^{-3} and a post-irradiation anneal at 1100 °C, a new set of dislocation loops is observed to form between the existing dislocation network. The second set of dislocation loops is thought to be formed from the collapse of platelets of vacancies.

PHIL. MAG., vol. 12, Dec. 1966

5. In-Pile Densification Characteristics of Uranium Dioxide, *W. R. Yario, S. T. Zegler, T. E. Hollowell, S. D. Harkness, M. G. Andrews, R. N. Duncan (CE)*

Combustion Engineering and Kraftwerk Union have pursued extensive irradiation test programs to establish the effect of fuel microstructure on fuel dimensional stability due to in-reactor densification of cold pressed and sintered UO_2. Data were also obtained to determine the relationship between UO_2 pellet dimensional changes and overall fuel volume changes.

An understanding of the densification phenomenon mechanisms is important to correlate various fuel microstructural features with in-reactor fuel densification data obtained from irradiation tests. At least two mechanisms for the densification process appear possible. The first[1] is an extension of classical sintering theory[2] in an environment where atomic diffusivity is enhanced. Although Marlowe[1] has been able to develop good correlation between in-pile and out-of-pile results with this approach, there appears to be reason for reservation. One is that in-pile, the pores would have to continue to act as point defect sources in an environment highly supersaturated with point defects. It is now well established that, under similar circumstances, pores in metals become point defect sinks. Another is that the grain growth expected during a high-temperature sinter is more than that expected at low temperatures in-reactor, since the classical grain growth processes would not be expected to operate when the grain boundaries are serving as sinks for a large flux of point defects. Experimentally, no measurable changes in grain size were observed during in-pile densification of unstable fuels at low fuel rod power levels.

The second mechanism involves fission-spike overlap of a pore. This process creates an amorphous layer of atoms on the surface of the pore, thereby causing a locally high concentration of vacant lattice sites which enables the pore to act as a source rather than a sink for vacancies for a short time. Mathematical developments of this mechanism[3] indicate that densification should increase with increasing fission rate, decreasing pore size, and decreasing grain size. Qualitatively, these are the same predictions of the in-reactor diffusion-enhanced classical sintering model. The adoption of a spike overlap model does not preclude the use of a thermal sintering test to assess in-pile densification characteristics.

A CE/KWU experimental fuel irradiation program on fuel densification has been carried out in the Battelle Research Reactor (BRR). The linear heat ratings ranged from 2.7 to 4.4 kW/ft, which corresponds to calculated fuel centerline temperatures of from 1070 to 1535°F (577 to 835°C). The fuel types included microstructures predicted to be both stable and unstable. The unstable

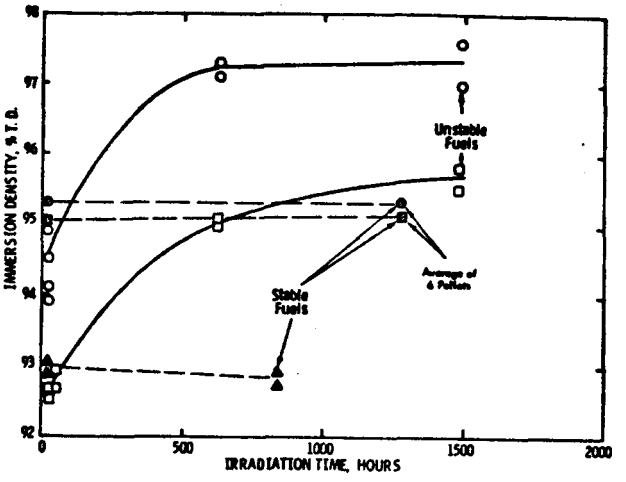


Fig. 1. UO_2 pellet density vs irradiation time for stable and unstable fuel (2.8 to 4.6 kW/ft).

microstructures were fine grained (<6 μm) and contained a large fraction of their pore volume in pores <4 μm in diameter. The stable fuel microstrucutres included a combination of large grain size and/or large pore size. The kinetics of densification were measured up to 1500 h. As shown in Fig. 1, the unstable fuels densified rapidly, while the stable fuels showed little change in density. Among the unstable fuels, a general trend toward increased densification with lower initial density was noted.

Diameters were measured directly on several pellets that had remained sufficiently intact to allow accurate measurement. The average ratio of diametral-to-volume change for six pellets was 0.29 or very near the 1/3 value expected if densification is isotropic.

The results of these CE/KWU experiments indicate that UO_2 fuel microstructure plays a dominant role in the kinetics and extent of in-reactor densification. Stable fuels can be produced by process control to obtain a UO_2 pellet microstructure with a large fraction of the pore volume being large pores. Both CE and KWU fuel fabrication processes yield UO_2 pellet fuels which are stable with respect to the in-reactor densification phenomenon.

1. M. O. MARLOWE, "In-Reactor Densification Behavior of UO_2," NEDO-12440 (July 1973).

2. R. L. COBLE, "Sintering Crystalline Solids—Intermediate and Final State Diffusion Models," *J. Appl. Phys.*, 32, 787 (1961).

3. H. ASSMANN and H. STEHLE, "Verdichtungseffekte in Gesintertem UO_2," *Reaktortagung, Karlsruhe 1973, Tagungsberight*, p. 409.

Blank Page

IV. QUALITY ASSURANCE ON FUEL, RODS, AND BUNDLES

Blank Page

4. Educational Needs in the Nuclear Fuel Industry—A Viewpoint, *Lamar P. Bupp (Exxon Nucl-Bellevue), Philip L. Farnsworth (Exxon Nucl-Richland), Chih H. Wang (Ore State Univ)*

This paper reviews five years of experience within the nuclear fuel cycle industries; specifically the nuclear fuel supply business. Attention is centered on technological areas where industry problems have occurred. The review looks at such issues as:

1. quality assurance and quality control challenges in supplying dependable nuclear fuel.
2. hydriding of zirconium cladding and the resultant problems this has caused in fuel performance
3. uranium oxide densification with its resultant impact on regulation and licensing for nuclear fuel supply
4. the emerging problem of pellet-to-cladding interaction and the limits this places on nuclear fuel power cycling.

These technological problem areas have required unique actions in the industry in establishing research and technology bases by which to deal with matters of safety and licensing. Cooperative research and development programs have emerged on fuel densification, high-temperature properties of zirconium cladding, and special tests related to power cycling.

Certainly, nuclear fuel industry problem areas serve as one strong basis for needs in academic training. When one compares materials problems in the nuclear fuel industry with materials problems in the nuclear steam supply system industry, some common concerns are evident and materials expertise overlap; i.e., stress corrosion cracking, weld closure integrity, mechanical/thermal stress-strain considerations, etc. Arguments for materials-based training in nuclear engineering, therefore, are broader than just the nuclear fuel cycle industry, as such.

It is vital to recognize that unique features of any nuclear power systems materials environment play an important role in the problem areas cited. Primary and secondary irradiation effects on material behavior are of paramount importance; particularly in long-term structural failures. The materials expert cannot deal in classic properties alone; he must have a concurrent appreciation for the fundamental nature of the fission process. This point is often overlooked in structuring curricula for the nuclear engineer. Moreover, this point is sometimes the cause for technical mistakes in interpreting materials behavior in the nuclear industry.

To provide specific examples, the fuel materials viewpoint derived from fuel performance problems the past five years in the nuclear industry may be cross-compared with Oregon State University's program on training nuclear engineering technologists. Oregon State University is one of the institutions that has pioneered in curricula applicable not only to nuclear engineering as a basis for advanced degrees in highly specialized fields, but also provides for BS graduates to have broader training in such fields as materials technology and the intricacies of applying materials considerations in this Nation's nuclear regulatory process.

The paper presents a viewpoint that augurs for heavier involvement in nuclear engineering training of disciplines associated with materials, metallurgy, ceramics, mechanical/thermal phenomena, and vagaries such as stress corrosion cracking and other similar-problem-related phenomena.

ANS TRANS., vol. 23, 1976

32809 (WCAP-7808(Vol.2)) PERFORMANCE OF ZIRCALOY CLAD FUEL RODS DURING A SIMULATED LOSS-OF-COOLANT ACCIDENT: MULTI-ROD BURST TESTS. VOLUME II. ANALYSIS OF RESULTS. Caso, C. L. (Westinghouse Electric Corp., Pittsburgh, Pa.). Dec 1971. 77p.

Based on the results of the Multi-Rod Burst Test (MRBT), peak clad temperature increases due to changes in fuel rod geometry in Westinghouse PWR cores following a loss-of-coolant accident are not expected to exceed 100°F. Analysis of the geometry distortion observed in the Multi-Rod Burst Test (MRBT) indicates that the fuel rods burst in a staggered manner thus precluding the occurrence of an extensive flow area blockage in the core during a loss of coolant accident. The regression line of the values of the maximum average (assembly-wise) flow area blockage on the heating rate and the initial pressure shows that the highest flow area blockage is 50 percent and occurs at low heating rates (<5°F/sec) and low pressure (200 psi). The corresponding assembly flow redistribution at the blockage location results in a reduction of 25 percent in the assembly mass flow rate when compared to the no blockage case. The effect of flow redistribution and rod geometry distortion on the clad temperature transient is determined by analyzing the three-dimensional temperature distribution in the hottest fuel rod. The reduction in coolant flow in the channels surrounding this rod is accounted for by correspondingly adjusting the rod surface convective heat transfer coefficients. No heat transfer is considered for the portion of the fuel rod in contact with the adjacent rods. The analysis of the temperature transient of a PWR reactor fuel rod with a no blockage peak clad temperature of 2300°F, yields an increase of 70°F in peak clad temperature when the worst rod geometry observed in the MRBT is considered. (auth)

NSA, vol. 26, 1972

2. Nuclear Fuel Performance in Oyster Creek, *B. H. Cherry, R. F. Denning (GPU Serv Corp), J. L. Sullivan (Jersey Central P&L), V. P. Zodiaco (GPU Serv Corp)*

This paper analyzes the nuclear fuel behavior at Oyster Creek. The reactor, a GE BWR rated at 1930 MW(th), began commercial operation in December 1969. It is now in its third cycle of operation. During this period, six different nuclear fuel designs have been irradiated. Variations between the fuel types include pellet size and geometry, pellet density, cladding thickness and properties, assembly mechanical structure, inclusion of burnable poison rods, and use of PuO_2-UO_2 fuels. Fuels fabricated over these years also vary in such critical areas as pellet impurity specifications and general level of quality assurance.

The fundamental assessment of fuel behavior is gained by wet sipping procedures. The Zircaloy channel enclosing the BWR assembly allows for in-core isolation of a particular assembly, and thereby permits measurement of activity release indicative of perforated rods. This operation was performed on all 560 assemblies during each outage and requires ~92 h. The paper develops the sipping-generated statistics by computing correlations with design factors, exposure level, core location, and other operating variables. Bundle failure (activity release to the coolant) is due to only a few perforated rods among the 49 composing the 7 × 7 assembly. Fuel rods successfully operated at percentages >99.5, even for the earliest cycles. The relationship between assembly and rod failure frequency indicates some positive correlation in fuel rod behavior within an assembly. The total number of perforated rods (based on off-gas analysis) would have resulted in a larger number of assembly failures if these rods were distributed with mathematical independence. The observed clustering of failures is felt due to similarity in the operating environment within an assembly rather than causal failure interaction mechanisms between rods.

With the exception of outages, the condition of fuel in the core must be assessed through off-gas analysis. A calculationally consistent summary of the Oyster Creek data is presented in terms of core average exposure, absolute time, reactor power level, and operating procedures. The isotopic composition of the gross activity is tracked through each of the cycles. This information, when analyzed in terms of isotopic half-life, is related to the sources of activity and types of failures implied.

Two fuel rod evaluation programs are discussed. At the end of cycle 1A (core average exposure—7500 MWd/ST), ~30 assemblies received rod by rod nondestructive evaluations as part of an assembly reconstitution program. The procedure takes advantage of the low rod failure rate and the mechanical disassembly feature of the assembly. Fuel rod swapping yields sound bundles which can be immediately reinserted in the core or stored for future cycles. A second program measured evidence of pellet column gapping in discharged fuel rods using a precise gamma scanning technique capable of identifying gaps as small as $\frac{1}{16}$ in. The paper discusses the observations and the implication to axial power spiking.

ANS TRANS., vol. 16, 1973

24027 DEVELOPMENT OF ZIRCALOY-4. J. N. Chirigos, S. Kass, W. W. Kirk, G. J. Salvaggio (Westinghouse Electric Corp., Pittsburgh). p.19-55 of "Fuel Element Fabrication with Special Emphasis on Cladding Materials. Volume 1." London and New York, Academic Press, 1961. (In English)

Although Zircaloy-2 has proven to be a desirable structural material for use in water-cooled reactors, research has resulted in the development of a new alloy named Zircaloy-4 which has lower hydrogen pick-up in water corrosion tests than does Zircaloy-2. The composition of Zircaloy-4 (1.10–1.70% tin, 0.12–0.18% iron, 0.05–0.15% chromium, remainder zirconium) is basically that of Zircaloy-2 except that no nickel is added. The corrosion behavior in high temperature water and steam, the physical and mechanical properties, the irradiation behavior, and fabrication of Zircaloy-4 is reviewed and these properties are compared with those of Zircaloy-2. In general, Zircaloy-2 and Zircaloy-4 are very similar in behavior except that during hot water corrosion Zircaloy-4 absorbs less hydrogen than does Zircaloy-2. Preliminary results indicate that during steam corrosion testing at 400°C the hydrogen pick-up is also low, but that at 454°C the oxidation rate and hydrogen absorption are higher for Zircaloy-4 than for Zircaloy-2. (auth)

NSA, vol. 15, 1961

2. Some Elements of an Effective Nuclear Fuel Quality Assurance Program, *Ronald R. Cost (W-NES)*

The introduction of Appendix B to 10CFR50 and the subsequent development of the family of ANSI standards identified the need for formal quality assurance programs that would provide additional confidence that design, procurement, and manufacturing activities associated with nuclear fuel assemblies and associated reactor components would result in hardware that would perform satisfactorily in service.

Since the introduction of 10CFR50 Appendix B, significant efforts have been devoted to the task of developing those more formal systems and organizations that would demonstrate and provide visibility of compliance. Among these was the restructuring of the Westinghouse Nuclear Fuel Division activities to consolidate and provide a focal point for all quality-related functions and activities within a product assurance discipline, while all disciplines throughout the Division were specifically charged with responsibility for developing, documenting, and implementing appropriate controls and quality assurance measures. This structuring provided organizational independence that satisfied 10CFR50 Appendix B and appropriate ANSI standards as applied to the manufacture of nuclear fuel and related components.

Specific elements of the Westinghouse Nuclear Fuel Division Quality Assurance Program are discussed and include the following:

1. *Design Control*: The Nuclear Fuel Division Engineering Department is responsible for mechanical, nuclear, and thermal-hydraulic design, development, and prototype testing. The safety and performance requirements to be met by the fuel assemblies and core components are established jointly by the Nuclear Fuel Division and the Pressurized Water Reactor Systems Division. The division of responsibility for design activities and the methods and procedures used are documented in an Engineering Operations Manual. The procedures in this manual are reviewed and approved by the Product Assurance Department.

The adequacy of design elements to satisfy design bases and regulatory requirements is evaluated through verification methods such as formal design reviews, verification of calculations by independent reviews, simulated and actual environmental condition testing. In each of these verification methods, the Product Assurance Department role is specified. Design requirements are ultimately defined in drawings, material and process specifications that are reviewed and approved by Quality Control which is within the Product Assurance Department.

2. *Supplier Qualification Program*: Suppliers of materials, services, or components must be qualified and acceptable to Quality Control. Suppliers attain qualification status through Quality Control evaluation of performance. New suppliers may receive orders for "qualification quantities" and are required to demonstrate compliance to appropriate quality criteria. Quality Control follows the suppliers' performance on these orders by surveillance, detailed receiving inspection, and ultimately an in-depth audit for compliance to appropriate quality criteria.

Once having been placed on a qualified suppliers list, qualification is maintained by evaluation of performance on subsequent orders. Assessment of continued qualification is made by surveillance, receiving inspection, and by annual compliance audits.

3. *Personnel Training and Qualification*: The Westinghouse NFD Quality Assurance program provides for indoctrination and training of personnel performing activities affecting quality. Each manager is charged with the responsibility for establishing requirements such as education, training, and experience for personnel reporting to him. He is further required to provide appropriate indoctrination, orientation, and training to ensure a high level of skills and performance. The training and qualification of personnel who perform inspection, examination, and testing activities that assure the quality of material, components, subassemblies, and assemblies is carefully planned and implemented.

4. *Audits*: The Product Assurance Department is responsible for setting policy, review, and approval of procedures describing quality-related activities. A comprehensive system of planned internal audits is conducted to verify compliance with these procedures. Audits are designed to assure coverage of all quality-related departments, areas, and activities at least once per year with appropriate followup by audit chairmen.

Audit teams are chaired by Product Assurance auditors who meet ANSI 45.2.12 requirements for training and experience. A broad distribution of audit reports assures management awareness.

The results of this concerted effort are evidenced by a recent approval by the Nuclear Regulatory Commission of the Westinghouse Quality Assurance Program Plan as an acceptable program for design, procurement, and manufacture of nuclear fuel and associated reactor components.

ANS TRANS., vol. 22, 1975

Quality Analysis in Pressurized Water Reactor Fuel, *J. F. Darolles (Framatome–France)*

INTRODUCTION

In this document we describe an integrated system which has been set up to administrate and analyze the quality. This system is in actual operation in our Fuel Division.

Basically, we must keep in mind the specific aspect of PWR fuel to understand the necessity of such a system and the proper methods that have to be used. The specific aspect lies in the component diversity and in their number.

Safety requirements are presently of high level and they will be higher in the future. These requirements impose on us a proper approach where rigor and intimate product knowledge must be a permanent obligation. Thus, it is a basic element of our quality assurance organization.

BASIC PRINCIPLES

Basic principles for quality analysis system, on one hand, are traceability, i.e., identification, location, and history of fuel components; and, on the other hand, quality evaluation during manufacturing.

Traceability

Our traceability system enables us to identify each fuel assembly and subassembly. It associates reference documents to them, such as certificates of compliance and inspection data.

The system enables locating each component in a fuel assembly or subassembly.

For example, a pellet tray with a given identification is used to load a given fuel rod which, itself, is loaded in a well-defined fuel assembly.

To fully understand the difficulty in setting up such a system, it must be kept in mind that a 900-MW(e) power plant core is constituted of some 43,000 fuel rods and some 10 million uranium dioxide pellets.

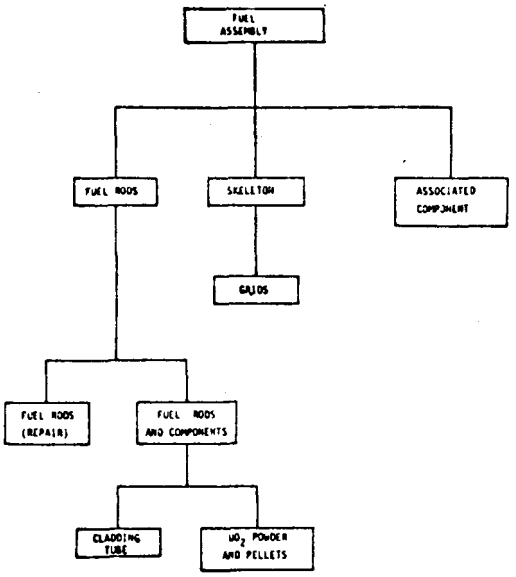


Fig. 1. Traceability file.

Each component, subassembly, and assembly must be located and identified. Therefore, anyone will understand how necessary it is to promote a system that uses the most advanced techniques to record and process data (route cards in adapted form for computer processing) and the most sophisticated computer programs for data computing.

The computer is used systematically for sorting and printing traceability files. (We have basically 800,000 traceability bits of information for a power plant core.)

Quality Evaluation

The objective of quality evaluation is to associate a quality coefficient to critical or major characteristics of a component, which enables rating with respect to specification limits and tolerances. This evaluation is performed for well-defined characteristics of each component–powder, UO_2 pellets, fuel rods, grids, fuel assemblies, etc. Combination of these quality coefficients allows determination of an integrated level of fuel assembly quality.

The two kinds of data transmitted in the frame of traceability and quality evaluation constitute the basis of our quality analysis system.

QUALITY ANALYSIS SYSTEM DESCRIPTION

The quality analysis system operates in the following areas:

Data Recording

The manufacturing organization must comply with the requirements of a data recording system; it must accept special modalities and follow a logic pattern for identification of operations and fuel components. Route cards take into account manufacturing organization features, computer technology, and general organization requirements. These route cards are divided into categories that correspond to the main components, which are the key points of the system (Fig. 1).

Data Transmission

We use a data transmission line, which is an extension of our computing system, between the Manufacturing Site and the Quality Analysis Center. By using such a setup, we can reply without delay to any question concerning quality.

Data Processing

Computer files are generated on tapes; they contain information coming from quality inspection, and are sorted according to given criteria which enable performing comparative studies (correlation analysis for example).

An examination of file completeness enables assurance of compliance of the final product to drawings and specifications. This is performed by making a step-by-step inspection of all file components.

Quality File Generation

Quality files, which are periodically printed, contain the data-computed results and give a synthesis of the quality, sorting out those available that are significant and representative as quality key points.

(Cont'd pg 125)

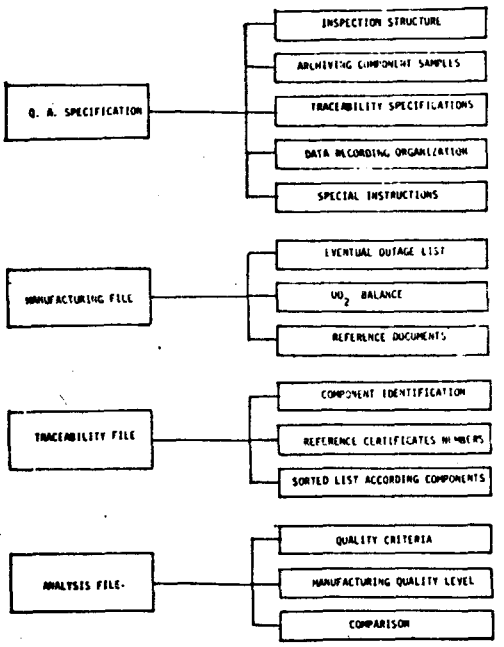


Fig. 2. Quality files.

QUALITY ANALYSIS IMPLICATION

The interest of such a system is to give full power to the information and use it as a technique for making decisions within the frame of knowledge of the product at all manufacturing stages.

Its implication may be noted particularly in the following areas:

During Manufacture

During manufacture of the product quality assurance enables piloting fabrication of all stages and preventing deviations by performance of statistical analyses of inspection data (variances and trend evaluations). Quality Assurance thus has the possibility of playing its part effectively and completely by promoting preventive measures rather than imposing corrective actions.

Quality Files

Quality files (Fig. 2) are constituted of and contain inspection results for all inspected characteristics. Completeness of these files is proof of product compliance to design requirements. In particular, we find the following information concerning fuel rods, which are an essential part of the fuel assembly.

Within the manufacturing file: (a) uranium weight contained in each fuel rod, (b) eventual defects or outage list, and (c) reference documents.

Within the traceability file: (a) fuel rod identification and location in fuel assemblies, (b) fuel rod component identification and references (spring, cladding tube, powder, pellets, plugs), (c) rational presentation allowing easy access to information (this would be impossible without the use of a computer).

Within quality evaluation file: (a) a list of design criteria for each component, (b) manufacturing results for the same criteria, and (c) a comparative evaluation between the two points above.

Feedback on Product Design

Quality file consultation is of first importance to the designer. In these documents he will find trends and manufacturing inspection results, and he will be able to remodel his criteria and define the performance of the product with a greater accuracy.

Data from Irradiated Fuel Assemblies

In performing correlation analyses, the data may be processed to find reasons for the given behavior of a fuel assembly and to determine eventual failure origin.

CONCLUSION

The various aspects of the quality analysis system described in this document have proved to us the importance of such a system toward fuel manufacturing. We have seen that it represents a basic feature of our Quality Assurance organization and gives its user a reliable means through computerized techniques to guarantee product compliance with respect to design requirements. It can be noted that quality analyses yield a data bank without which it would be impossible to draw any conclusions, despite the enormous amount of available information. Furthermore, it is a factor of economy for us by the level of security it imparts to decisions.

NUC. EN. MATUR., vol. 6, 1976

Quality Analysis in Pressurized Water Reactor Fuel, *J. F. Darolles (Framatome–France)*

INTRODUCTION

In this document we describe an integrated system which has been set up to administrate and analyze the quality. This system is in actual operation in our Fuel Division.

Basically, we must keep in mind the specific aspect of PWR fuel to understand the necessity of such a system and the proper methods that have to be used. The specific aspect lies in the component diversity and in their number.

Safety requirements are presently of high level and they will be higher in the future. These requirements impose on us a proper approach where rigor and intimate product knowledge must be a permanent obligation. Thus, it is a basic element of our quality assurance organization.

BASIC PRINCIPLES

Basic principles for quality analysis system, on one hand, are traceability, i.e., identification, location, and history of fuel components; and, on the other hand, quality evaluation during manufacturing.

Traceability

Our traceability system enables us to identify each fuel assembly and subassembly. It associates reference documents to them, such as certificates of compliance and inspection data.

The system enables locating each component in a fuel assembly or subassembly.

For example, a pellet tray with a given identification is used to load a given fuel rod which, itself, is loaded in a well-defined fuel assembly.

To fully understand the difficulty in setting up such a system, it must be kept in mind that a 900-MW(e) power plant core is constituted of some 43,000 fuel rods and some 10 million uranium dioxide pellets.

Each component, subassembly, and assembly must be located and identified. Therefore, anyone will understand how necessary it is to promote a system that uses the most advanced techniques to record and process data (route cards in adapted form for computer processing) and the most sophisticated computer programs for data computing.

The computer is used systematically for sorting and printing traceability files. (We have basically 800,000 traceability bits of information for a power plant core.)

Quality Evaluation

The objective of quality evaluation is to associate a quality coefficient to critical or major characteristics of a component, which enables rating with respect to specification limits and tolerances. This evaluation is performed for well-defined characteristics of each component–powder, UO_2 pellets, fuel rods, grids, fuel assemblies, etc. Combination of these quality coefficients allows determination of an integrated level of fuel assembly quality.

The two kinds of data transmitted in the frame of traceability and quality evaluation constitute the basis of our quality analysis system.

QUALITY ANALYSIS SYSTEM DESCRIPTION

The quality analysis system operates in the following areas:

Data Recording

The manufacturing organization must comply with the requirements of a data recording system; it must accept special modalities and follow a logic pattern for identification of operations and fuel components. Route cards take into account manufacturing organization features, computer technology, and general organization requirements. These route cards are divided into categories that correspond to the main components, which are the key points of the system (Fig. 1).

Data Transmission

We use a data transmission line, which is an extension of our computing system, between the Manufacturing Site and the Quality Analysis Center. By using such a setup, we can reply without delay to any question concerning quality.

Data Processing

Computer files are generated on tapes; they contain information coming from quality inspection, and are sorted according to given criteria which enable performing comparative studies (correlation analysis for example).

An examination of file completeness enables assurance of compliance of the final product to drawings and specifications. This is performed by making a step-by-step inspection of all file components.

Quality File Generation

Quality files, which are periodically printed, contain the data-computed results and give a synthesis of the quality, sorting out those available that are significant and representative as quality key points.

QUALITY ANALYSIS IMPLICATION

The interest of such a system is to give full power to the information and use it as a technique for making decisions within the frame of knowledge of the product at all manufacturing stages.

Its implication may be noted particularly in the following areas:

During Manufacture

During manufacture of the product quality assurance enables piloting fabrication of all stages and preventing deviations by performance of statistical analyses of in-

(Cont'd pg 127)

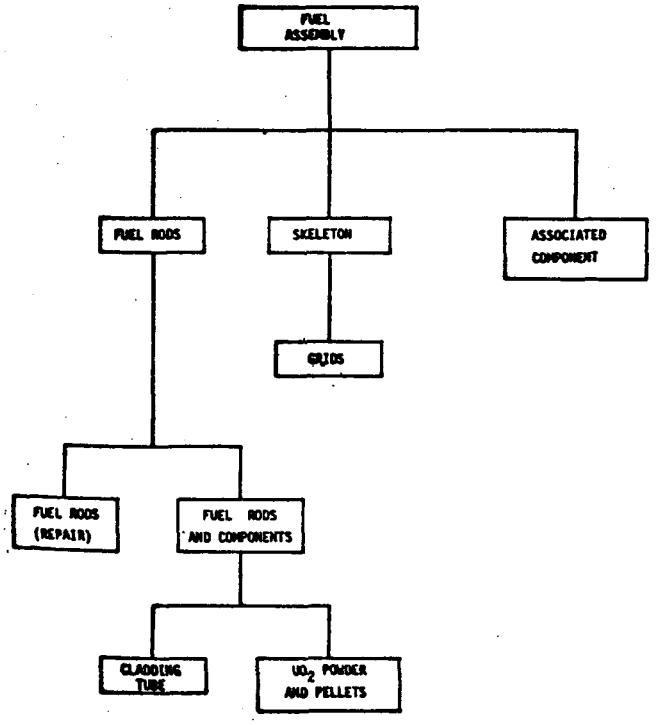


Fig. 1. Traceability file.

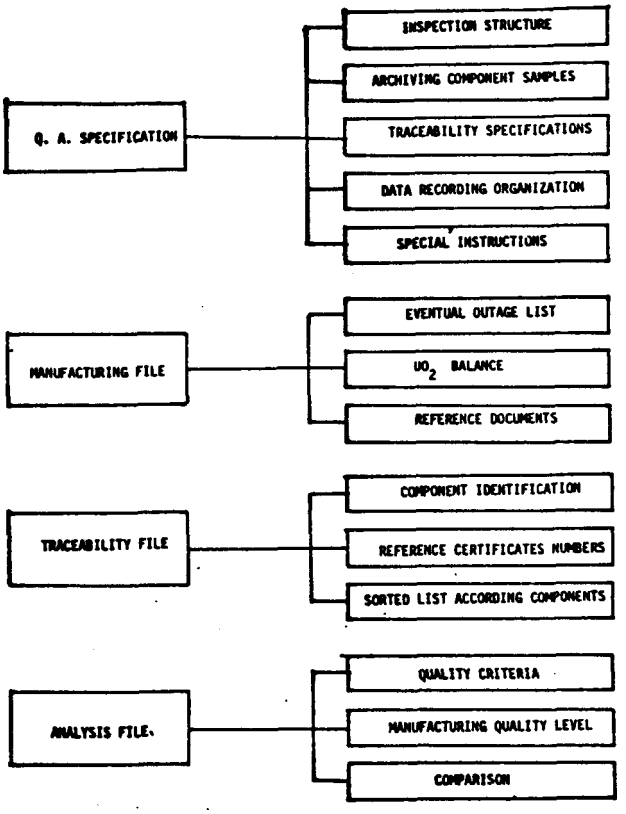


Fig. 2. Quality files.

spection data (variances and trend evaluations). Quality Assurance thus has the possibility of playing its part effectively and completely by promoting preventive measures rather than imposing corrective actions.

Quality Files

Quality files (Fig. 2) are constituted of and contain inspection results for all inspected characteristics. Completeness of these files is proof of product compliance to design requirements. In particular, we find the following information concerning fuel rods, which are an essential part of the fuel assembly.

Within the manufacturing file: (a) uranium weight contained in each fuel rod, (b) eventual defects or outage list, and (c) reference documents.

Within the traceability file: (a) fuel rod identification and location in fuel assemblies, (b) fuel rod component identification and references (spring, cladding tube, powder, pellets, plugs), (c) rational presentation allowing easy access to information (this would be impossible without the use of a computer).

Within quality evaluation file: (a) a list of design criteria for each component, (b) manufacturing results for the same criteria, and (c) a comparative evaluation between the two points above.

Feedback on Product Design

Quality file consultation is of first importance to the designer. In these documents he will find trends and manufacturing inspection results, and he will be able to remodel his criteria and define the performance of the product with a greater accuracy.

Data from Irradiated Fuel Assemblies

In performing correlation analyses, the data may be processed to find reasons for the given behavior of a fuel assembly and to determine eventual failure origin.

CONCLUSION

The various aspects of the quality analysis system described in this document have proved to us the importance of such a system toward fuel manufacturing. We have seen that it represents a basic feature of our Quality Assurance organization and gives its user a reliable means through computerized techniques to guarantee product compliance with respect to design requirements. It can be noted that quality analyses yield a data bank without which it would be impossible to draw any conclusions, despite the enormous amount of available information. Furthermore, it is a factor of economy for us by the level of security it imparts to decisions.

IAEA International Seminar on Nuclear Fuel Quality Assurance (Oslo, Norway, 24-28 May 1976)

The objectives of the seminar were to provide educational lectures on the basic concepts of Quality Assurance (QA) and Quality Control (QC) as applied to nuclear fuels and to review the current methods of applying these in practice. These goals were achieved.

On the basis of the presentations and discussions at the meeting recommendations for future Agency activities in this area have been prepared. These include expansion of present efforts, regional seminars in the developing countries, a central information service in QA and QC development activities, and an Advisory Group on HTR fuel QA development.

The widespread interest in the topic of QA for nuclear fuels was demonstrated by the presence of participants from 25 countries. Furthermore, there was very active discussion in all of the sessions, especially in response to the lead lectures with a basic educational purpose. The participants represented a remarkable range of interests including fuel manufacturers, users (utilities and their consultants), regulatory bodies, process development personnel, and technicians from developing countries, while the discussions covered topics from the general philosophy of QA to details of QC related to LWR, LWR recycle, FBR, and HTR fuels.

The need for more detailed regional seminars on this topic, aimed at assisting the developing countries, was emphasized by the presence (and discussion period questioning) of 16 participants from nine developing countries.

To summarize, the most important points raised during the seminar discussion periods were: -

— Vendor-utility interaction — the question of how much information should be provided by the fuel manufacturer (the vendor) to the utility which is purchasing fuel, and in what form and how early this information should be provided, was a recurring subject of discussion. Opinion was divided on these points with some participants happy with their present arrangements and others not. It appears that the information transfer is basically set out in the individual vendor-utility contracts — with some indirect influence from the regulatory bodies through their requirements on the utility. Any conflict that exists appears to come from those utilities and their QA consultants which desire more extensive control over the product they are buying. There is a natural reluctance on the part of vendors to permit this but there seems to be an increasing trend toward release of more information thus involving both parties in closer co-operation. The discussion and illumination of these problems fulfilled one of the objectives of this seminar.

  The close co-operation of vendor and utility can only benefit all parties — in particular since the utility has the best source of the real fuel performance data needed for feed-back to fuel design and QA evaluation.

(Cont'd pg 129)

- QA of fuel design — this area of fuel QA recurringly appeared in the discussions. There was general agreement that it was the weakest point in the QA scheme at present and efforts are needed to remedy this problem. No general statement of how to accomplish this was made.
  There are apparently two basic difficulties which confound attempts to move rapidly in design QA. One is the strong commercial significance of improved design and the other is the fact that much of the data on fuel failure is in the control of the utilities who have generally not been prepared for in-depth fuel examinations of a diagnostic nature. Nevertheless, attempts are under way to pool the utilities and vendors' information into data banks and connect fuel performance with design and specification. It is felt these efforts were encouraged and the significance of the problem highlighted through this seminar.

- Fuel Specifications — this topic was approached from two viewpoints. There was an obvious interest on the part of fuel purchasers (developing countries and utilities) in getting as much detail of the fuel specifications as possible. This information is, however, not normally made available except through individual buyer-seller contracts.

  On the other end of the scale, there were comments by the manufacturers that often specifications were set without proper regard for the latest technical information on fuel performance and for the realities of manufacturing processes and testing capabilities. This problem will be resolved when the fuel design activities are properly meshed with the full QA system.

- Fuel Failure and QA — the costs of fuel failures and the desire for increased safety were mentioned as strong justifications for fuel QA. programs. One author quoted the cost of each fuel failure as being US$45,000.- and another cited the cost for a one-day outage of a 500 MWe plant as US$100,000. However, these figures should be applied to fuel with care. Advantage is usually taken of any reactor shut down to carry out several tasks, thus reducing the cost to be assigned to any one cause. Further, the value of the quality assurance program in reducing fuel failures is difficult to separate from the design improvements resulting from R&D programs initiated outside the QA program. It should also be recognized that fuel failures may also result from improper reactor operation not related to the fuel QA.

"Meetings on atomic energy", vol. 8, n. 4, 1976

19730 (BAW–10035) FUEL ASSEMBLY STRESS AND DEFLECTION ANALYSIS FOR LOSS-OF-COOLANT ACCIDENT AND SEISMIC EXCITATION. (Nonproprietary Version of BAW–10008, Part 2, Rev. 1). DeMars, R. V.; Steinke, R. R. (Babcock and Wilcox Co., Lynchburg, Va.). Jan 1972. 72p.

The capability of the fuel assemblies to withstand the effects of a loss-of-coolant accident (LOCA) blowdown, the operational basis earthquake (OBE) and design basis earthquake (DBE), and the simultaneous occurrence of the DBE and LOCA are evaluated. This method of analysis is applicable to nuclear steam system contracts that specify the skirt-supported pressure vessel. Loads during the saturated and subcooled phases of blowdown following a loss-of-coolant accident were calculated. This LOCA loading is also applicable to all skirt-supported vessels. The seismic excitations will be determined for each reactor site. However, the description of reactor components, the analytical model, testing, and LOCA loadings is generally applicable. The maximum loads on the fuel assemblies were found to be below allowable limits, and the maximum deflections of the fuel assemblies were found to be less than those that could prevent the insertion of control rods or the flow of coolant through the core. Based on the results of this investigation, it is concluded that the fuel assemblies can withstand the loss-of-coolant accident and seismic excitation loadings individually, as well as a simultaneous LOCA and DBE, without exceeding allowable limits. (auth)

NSA, vol. 26, 1972

14844 UTILITY QUALITY ASSURANCE PROVISIONS FOR MIXED OXIDE FUEL. Denning, R. F. (General Public Utilities Service Corp., Parsippany, NJ). Nucl. Mater. Manage.; 11: No. 3, 150-179(Fal 1973).

From proceedings of the 14th annual meeting of the Institute of Nuclear Materials Management, Inc.; San Diego, CA (20 Jun 1973).

The scale of plutonium recycle should climb sharply in the next few years reaching a total of more than fifty reloads per year by 1980. Utilities involved will be responsible for the integrity of the mixed oxide fuel. A series of quality assurance considerations which differentiate this fuel type from enriched uranium designs are presented. The areas of design, manufacturing, quality control, and on-site procedures are handled. Particular attention is devoted to the particle size distribution of the PuO_2 agglomerates in the UO_2 matrix. A variety of corrections are discussed appropriate for transforming the results of alpha radiography (or equivalent techniques) into a volume particle distribution. 11 references.

NSA, vol. 29, 1974

33840 (IAEA-TA–527) FUEL ELEMENTS TESTING. Report to the Government of Argentina. Hess, E. G. (International Atomic Energy Agency, Vienna (Austria)). 24 Mar 1970. 5p. INIS.

Methods used in programs to advise on the planning of work and to specify the tests to be carried out on power reactor fuel elements of the UO_2-Zircaloy-type are described. The general philosophy and basic specifications of a pressurized water out-of-pile loop for testing full-size Atucha fuel elements and a program of metallurgical research on the influence of fabrication techniques on the properties of Zircaloy and related alloys including equipment for testing and evaluating Zircaloy properties are discussed. (INIS)

NSA, vol. 26, 1972

263405 International Atomic Energy Agency, Vienna (Austria). **Quality assurance and control in the manufacture of metalclad UO_2 reactor fuels.** Report of a panel on quality assurance and control in nuclear fuel manufacture organized by the IAEA and held in Vienna 4–7 November 1974. ISBN 92-0-155076-6. Vienna, IAEA, 1976. 66 p. Technical reports series, no. 173.

The International Atomic Energy Agency has carried out a programme since its earliest days that includes the collection and dissemination of information on nuclear fuels. Since the 1960 symposium on Fuel Element Fabrication with Special Emphasis on Cladding Materials there has been an average of one meeting a year reviewing some aspect of fuel fabrication technology. A recent meeting dealing with the fabrication of UO_2 fuels was the Study Group on the Facilities and Technology needed for Nuclear Fuel Manufacture, held in Grenoble in 1972 (Rep. IAEA–158). After that meeting it became apparent that the quality of fuel production was an important aspect that had received inadequate coverage so far, and the Panel on Quality Assurance and Control in Nuclear Fuel Manufacture was convened by the Agency in Vienna in November 1974. In the working papers and discussions at the Panel meeting the viewpoints of different countries and of various interested parties, such as manufacturers, reactor operators and government authorities, were presented.

NUCLEAR FUELS: fabrication: QUALITY ASSURANCE: nuclear fuels: QUALITY CONTROL: nuclear fuels: URANIUM DIOXIDE: nuclear fuels.

Atomindex, vol. 7, 1976

54098 (NEDO–10017) FIELD TESTING REQUIREMENTS FOR FUEL, CURTAINS AND CONTROL RODS. Jones, D. R.; Hersum, T. G. (General Electric Co., San Jose, Calif. Atomic Power Equipment Dept.). Jun 1969. 9p.

Analyses and experimental data are presented to show that the testing of nuclear properties of core components in the field is not necessary. This conclusion is substantiated by analyses which show that the present pattern of quality control measures on fuel, curtains, and control rods provides sufficient assurance that the fuel and control-worth conform to design for both value and uniformity. (auth)

NSA, vol.25, 1971

42524 EXPERIMENTAL DETERMINATION OF BURN UP AND HEAVY ISOTOPE CONTENT IN A BWR FUEL ASSEMBLY AND COMPARISON WITH BURN UP CALCULATIONS. Koch, L. (EURATOM, Karlsruhe, Ger.); Cricchio, A.; Ariema, A.; Paoletti, M.; Zaffiro, B. pp 543-6 of Reaktortagung, Bonn, 1971. Bonn; Deutsches Atomforum E. V. (1971).

From Reactor meeting; Bonn, Germany (30 Mar 1971). See CONF-710346.

Post-irradiation examinations were made in order to ascertain the capabilities of predictive calculations. A BWR fuel assembly was subjected to an extensive post-irradiation analysis, the results of which were compared with those obtained by different burnup codes. Especially calculations for burnup and heavy isotope distribution of individual rods of the assembly had to be verified. (INIS)

NSA, vol. 26, 1972

9177 NONDESTRUCTIVE TESTING OF NUCLEAR CERAMIC MATERIALS. McClung, R. W. (Oak Ridge National Lab., Tenn.). Amer. Ceram. Soc. Bull.; 49: 777-81(Sep 1970).

Nondestructive testing techniques can be used as aids to the ceramist and developer of fuel elements for the evaluation of properties of uranium, thorium, and plutonium compounds and components. Ultrasonic and eddy current techniques were used for measurement of physical properties of polycrystalline specimens. After formation into microspheres, microradiography was used as an aid to product development, process control, product acceptance, and postoperation evaluation. Measurements of radiation attenuation are useful as a monitor for in-line control for coating of the process particles. After assembly into spheres, infrared techniques can detect the nonbond between the fueled core and unfueled shell, and eddy current techniques detect the presence of flaws in the shell. After assembly into graphite-bonded fuel sticks or metal-clad fuel rods, qualitative imaging by radiography and quantitative measurement of radiation attenuation are useful for determination of variation in fuel content. (auth)

NSA, vol. 26, 1972

15522 (IAEA-145, pp 427-431) NONDESTRUCTIVE EXAMINATION OF NUCLEAR FUEL PELLETS IN THE U. S. A. McClung, R. W. (Oak Ridge National Lab., TN). 1972.

From meeting on NDT for reactor core components and pressure vessels; Vienna, Austria (29 Nov 1971).

Several nondestructive evaluations are made of uranium- and plutonium-bearing fuels to assure conformance to specifications. These include visual examination with comparison to physical standards and measurement of dimensions, density, and fuel content. In addition to these that are commonly used, thermal techniques have been studied for density measurement, gamma attenuation has been used for measurement of fuel concentration, and both electron microprobe and autoradiographic techniques have been used to measure plutonium distribution in fuel pellets containing mixtures of plutonium and uranium oxides. (auth)

NSA, vol. 29, 1974

3. Performance of B&W PWR Fuel, *J. T. Mayer, M. H. Montgomery (B&W), invited*

There are presently six operating B&W PWRs in the U.S., all of which utilize the Mark B 15 × 15 fuel design. The lead plant (Oconee-1) was refueled in late 1974 after a successful first cycle of operation. An extensive examination has shown the fuel to be in excellent condition.

TABLE I

Current B&W Fuel Performance

Reactor	Rods Irradiated to Date	Maximum Assembly Burnup (2-1-76) (MWd/MTM)	Fuel Integrity (%)
Oconee-1	49 504	20 000	>99.99
Oconee-2	36 816	16 000	>99.99
Oconee-3	36 816	12 500	100
Three Mile Island	36 816	17 500	>99.99
Arkansas Nuclear One	36 816	12 500	>99.99
Rancho Seco	36 816	6 300	100

This paper will discuss fuel performance at all plants, with special emphasis on the results of the Oconee-1 examination.

The excellent fuel performance at all six plants is outlined in Table I. This shows the fuel integrity level based on equilibrium ^{131}I concentration in the primary coolant over the last quarter of 1975. Note that all plants are above the 99.99% integrity level, and two of the plants show activity levels consistent with zero-defect fuel. It should also be noted that fuel performance has generally improved with each successive plant that has come on line.

The first of a three-cycle nondestructive examination of Oconee-1 fuel was completed in 1975. Thirty-six fuel assemblies were examined. The visual appearance of the fuel was quite good, showing no evidence of significant rod bowing or grid distortion. A large amount of data were taken to characterize fuel rod bow,[1] which was <14 mil in 95% of the measurements. Fuel assembly bow was found to average $<\frac{1}{4}$ in. Rod growth averaged 0.2%, while assembly growth was somewhat less. Very little change was found in the preloads of either the holddown springs or the grid springs. Crud samples, composed primarily of NiO and $NiFe_2O_4$, were taken from several assemblies. Two different thickness measurement techniques resulted in fairly consistent values between 0.3 and 1.0 mil.

In addition to the nondestructive program, a three-cycle destructive test phase has also begun. Following cycle 1 refueling, one of the Oconee-1 discharged assemblies was shipped to the B&W hot cells for destructive examination. This particular assembly was extensively pre-characterized when it was manufactured, so that accurate beginning-of-life values would be available to compare with irradiated values. Nineteen rods were removed from the assembly and have undergone extensive examination to characterize rod length, fuel column length, burnup profile, diameter profile, and cladding thickness. Fission gas analysis, rod sectioning, metallography, and physical/chemical testing have also been done. Preliminary results from the hot cell examination show good agreement with the nondestructive test phase results, and the fuel design and operating models.

In summary, measurements to date have confirmed that the fuel assembly is conservatively designed and operating as expected.

1. R. A. KING, J. T. MAYER, and J. C. SPAZIANI, "In-Reactor Fuel Rod Bowing Characteristics," *Trans. Am. Nucl. Soc.*, 23, 150 (1976).

ANS TRANS., vol. 23, 1976

263328 Meyer, H.J.; Prestel, W. (Maschinenfabrik Augsburg-Nuernberg (M.A.N.) A.G., Nuernberg (F.R. Germany). Abt. fuer Qualitaetswesen). **Ultrasonic inspection of thin wall welds with quantification and recording represented on the example of zirconium fuel cans.** (In German). Ultraschallpruefung duennwandiger Naehte mit Quantifizierung und Registrierung dargestellt am Beispiel von Zirkonium-Brennelementkaesten. *Materialpruefung.* (Mar 1976). v. 18(3) p. 76-81. 9 figs.; 10 refs. [Lecture meeting of the Deutsche Gesellschaft fuer Zerstoerungsfreie Pruefung, Berlin, F.R. Germany, 6 May 1975.]

Following extensive comparison investigations between radiographic and ultrasonic inspections and many years development work of mechanized ultrasonic equipment for zirconium fuel cans, it is possible to carry out a 100% volumetric ultrasonic inspection on the two channel longitudinal welds of 2 mm wall thickness and less with the specified failure traceability. (orig.).

FUEL CANS: ultrasonic testing; WELDED JOINTS: ultrasonic testing.

Atomindex, vol. 7, 1976

3. Quality Assurance for Fuel Supply, *N. Mostin, P. Libotte (Belgonucleaíre)*

Belgonucleaíre includes three operational divisions: the Engineering division, which is not concerned by this paper; the Fuel Engineering division (FE), whose principal aim is to provide all services related to the design and supply of reactor cores; the Fuel Production division (FP), which is mainly concerned with the manufacturing of plutonium oxide fuel pins and subassemblies for thermal and fast reactors, and is located at about 50 miles from the headquarters.

This paper describes the QA organization of Belgonucleaíre in the field of fuel supply.

Belgonucleaíre offers these types of services:

1. Whole core and fuel supply in which both divisions are involved: in this case the FE division is the responsible supplier and the FP division, responsible for the quality of the product and its conformity to specifications, works as subcontractor to the FE division.

2. Fuel manufactured by the FP division for companies other than Belgonucleaíre or fuel engineering services by the FE division: in these cases, both divisions work independently of each other.

Consequently, each division has its own quality assurance service.

The QA organization of the FE division takes into account the following requirements: (a) 10CFR50 Appendix B regulations, as a guideline; (b) relatively small size of the Fuel Engineering division; and (c) flexibility to meet the requirements of the different national regulations.

When the Engineering work is done in association with partners, the work performed by one partner is checked by the other. Typical cases of work are SENA (PWR) and SNR (FBR). The nuclear, thermodynamic, and mechanical design is checked by another individual reporting to the QA manager, the specifications and drawings are reviewed, and must be approved by the QA manager before release for fabrication.

To assure the reliability of its fuel supply, Belgonucleaíre performs a qualification program including extensive design works as well as in- and out-of-pile experiments.

Fuel fabrication is audited by the FE quality assurance. Basically, and after qualification, the QA organization of the manufacturers is maintained as such and the FE QA homogenizes the quality level and presentation of the records.

The QA service of the FP division has the function of assuring the quality of products delivered to the customer, and reports directly to the manager of the FP division.

The QA service channels the contacts with the QA of the customers (among others, the FE division), gives agreement on the specifications, and coordinates the actions related to the Materials Review Board, which includes two sections—quality control and quality audit—to which the following tasks are assigned, respectively:

1. setting up the control plan
2. approval of all fabrication, measurement, and analysis procedures,
3. sub-supplier qualification
4. interpretation of the results of the inspection to verify either conformity or discrepancy of the product quality
5. establishing of the control certificates

and

1. audit of all production procedures and reception of feed material and components at subcontractor's plant
2. audit of all the production procedures (fabrication, measurement, and analysis) at FP division plant.

Quality Assurance organization significantly improves reliability of the work but it costs time and money; and a too-heavy system would lead to an unacceptable decrease of productivity. Therefore, another task of the QA manager consists of proposing a sound compromise between quality of the product (soft and hardware) and restraint of the procedures.

ANS TRANS., vol. 22, 1975

19731 (ORNL-TM-3626) EFFECT OF IRRADIATION ON THE FAILURE OF ZIRCALOY-CLAD FUEL RODS. Osborne, M. F.; Parker, G. W. (Oak Ridge National Lab., Tenn.). Jan 1972. Contract W-7405-eng-26. 36p. Dep. NTIS.

A loss-of-coolant accident in a water-cooled power reactor probably would cause a rapid temperature excursion in the core. A program to investigate the behavior of the Zircaloy-clad UO_2 fuel rods in such a thermal transient was conducted. The results of fuel rod failure experiments in steam atmosphere on both irradiated and unirradiated fuel rods are discussed. The results were compared with the mechanical properties of Zircaloy, with related work by other experimenters, and with a failure model. Cladding expansions of 15 to 70% were observed with rods that ruptured in the temperature range 1450 to 2600°F with thermal pressures of 50 to 1000 psig. For irradiated rods at fast neutron exposures of $\sim 1 \times 10^{21}$ nvt (>1.0 MeV) the average expansion was about 70% of the expansion in unirradiated rods. These data correlate reasonably well with a failure model based on secondary

NSA, vol. 26, 1972

QUALITY ASSURANCE IN THE MANUFACTURE OF TUBE FOR NUCLEAR SYSTEMS

R.W. PETHEN, *J.C. GUEST and **D.M. MARSH
Round Oak Steel Works., P.O. Box No. 3, Brierley Hill, West Midlands, U.K.
*Tube Investments Ltd., STD Development Engineering Dept., The Airport, Walsall, U.K.
**TI Research Laboratories, Hinxton Hall, Nr. Saffron Walden, Essex, U.K.

SUMMARY

This paper discusses the quality assurance systems in use by the Steel Tube Division of Tube Investments Ltd. It is shown that such systems not only require careful consideration before manufacture but also need massive backing in both applied and basic research areas. Two examples of this back-up are discussed in some detail. The first is a study of the significance of various defects in practical operating conditions. The second is a fundamental exploration of the behaviour of ultrasound in the ultrasonic test both by visualization and computer methods.

NUC. EN. MATUR., vol. 6, 1976

10255 CONTROL OF A NUCLEAR POWER PLANT BEFORE REACTOR OPERATION. Potemans, M. (Kerncentrale, Deol, Belg.); Gilly, L. Electricite; No. 159, 29-38(1974). (In French).

The safety, reliability, and profitability of nuclear power plants receive particular attention because of the harmful effects of a potential accident, their high unit power, and their use as base units for the production of electricity. That is why the quality controls during fabrication and the preoperational tests and the tests at start-up should be and are much more rigid than for conventional power plants. An attempt is made to describe briefly these controls and tests which are applied to nuclear power plants of the pressurized water type such as those at Doel and Tihange. Quality control during fabrication, preoperational tests, and the commissioning of the power plant are discussed. Although the majority of the components and systems of a nuclear power plant do not differ in principle from those of a conventional power plant, some of them are specific to nuclear power plants. These are, primarily, the sealed pressure vessel, the control rods, the reactor vessel, and the processing of radioactive wastes. Only those aspects peculiar to nuclear power plants which are subjected to particular tests and to more severe controls than their conventional outerparts are considered. (tr-auth)

NSA, vol. 31, 1975

13640 QUALITY: COMMON SENSE. Rahiser, R. H. pp 76.1-76.4 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004.

The interpretation of the results in terms of statistical confidence levels and how all these functions relate in meeting the needs of the customer in terms of performance are analyzed. The quality system as it relates to these functions is described along with tools available today for implementing the program. The requirements placed upon the product reflects the importance of the characteristic and the use of the results by the knowledge and reliability of the information obtained. (auth)

NSA, vol. 32, 1975

9178 CONTACT MICRORADIOGRAPHIC INSPECTION OF COATED NUCLEAR FUEL PARTICLES. Rooney, J. (United Kingdom Atomic Energy Authority, Springfields, Eng.). Non-Destruct. Test.: 4: No. 5, 338-43(Oct 1971).

The application of microradiographic techniques to the examination of coated nuclear fuel particles and the development of two separate methods are described. The first method uses commercial x-ray film and gives a high contrast image of limited definition by which large numbers of particles may be inspected. The second method uses high resolution scientific plates giving a low contrast image of very high definition but it has the disadvantage that the plates require a long exposure. (auth)

NSA, vol. 26, 1972

16529 ZIRCALOY CLADDING TUBES: MANUFACTURING TECHNIQUES AND ACHIEVABLE QUALITY: A TUBE MANUFACTURER'S VIEW. Rose, R.; Lunde, K.; Aas, S. (Institutt for Atomenergi, Kjeller (Norway)). Nucl. Eng. Des.; 33: No. 2, 219-229(Sep 1975).

The behavior of the Zircaloy fuel cladding in water-cooled power reactors depends on a variety of properties determined during the cladding tube fabrication process. This does not necessarily imply that variations in the fabrication process will determine the fuel pin integrity, although the considerable number of tube specification changes introduced over the last few years may well be interpreted in that way. (NL)

NSA, vol. 33, 1976

15013 NUCLEAR FUEL FABRICATION AND QUALITY ASSURANCE. Smiley, S. H. Nucl. Mater. Manage.; 11: No. 3, 463-472(Fal 1973).

From proceedings of the 14th annual meeting of the Institute of Nuclear Materials Management, Inc.; San Diego, CA (20 Jun 1973).

The AEC's quality assurance requirements are described for nuclear fuel design and fabrication, and for nuclear power plant design, construction, and operation. (DCC)

NSA, vol. 29, 1974

32808 (WCAP-7808(Vol.1)) PERFORMANCE OF ZIRCALOY CLAD FUEL RODS DURING A SIMULATED LOSS-OF-COOLANT ACCIDENT: MULTI-ROD BURST TESTS. VOLUME I. TEST SET-UP AND RESULTS. Schreiber, R. E.; Lange, R. A.; McKown, D. L. (Westinghouse Electric Corp., Pittsburgh, Pa.). Dec 1971. 125p.

Twelve transient burst tests were conducted. Each test was composed of 64 Zircaloy-4 clad rods (three feet long by 0.422 inch O.D.) in a square array and standard pitch (0.563 inch). ZrO_2 coated (nominal 5 mils), resistance heated, and filled with Al_2O_3 pellets. Fourteen rods in the center 4 × 4 were pressurized; two were unpressurized and unheated to represent rod cluster control (RCC)thimbles. The two surrounding rows of guard rods, heated but unpressurized, provide the appropriate thermal environment. The bursts produced during the transient were randomly distributed in time and location, thus demonstrating that acceptable flow restriction would result. Detailed analysis of flow blockage and correlation with temperature, heating rate, and pressure are summarized in Volume II. (auth)

NSA, vol. 26, 1972

31928 NONDESTRUCTIVE ANALYSIS OF FUEL PINS. Stepan, I. E.; Allard, N. F.; Suter, C. R. (to United States Atomic Energy Commission). U. S. Patent 3,755,675. 28 Aug 1973. Filed date 3 Nov 1972. 8p.

A method and a correspondingly adapted facility are described for the nondestructive analysis of the concentration of fuel and poison in a nuclear reactor fuel pin. The concentrations of fuel and poison in successive sections along the entire length of the fuel pin are determined by measuring the reactivity of a thermal reactor as each successive small section of the fuel pin is exposed to the neutron flux of the reactor core and comparing the measured reactivity with the reactivities measured for standard fuel pins having various known concentrations. Only a small section of the length of the fuel pin is exposed to the neutron flux at any one time while the remainder of the fuel pin is shielded from the neutron flux. In order to expose only a small section at any one time, a boron-10-lined dry traverse tube is passed through the test region within the core of a low-power thermal nuclear reactor which has a very high fuel sensitivity. A narrow window in the boron-10 lining is positioned at the core center line. The fuel pins are then systematically traversed through the tube past the narrow window such that successive small sections along the length of the fuel pin are exposed to the neutron flux which passes through the narrow window. (Official Gazette)

NSA, vol. 28, 1973

FABRICATION OF VIBRASOL FUEL RODS

P.F. SENS
Interfuel B.V.,
Amsterdam

J.B.W. KANIJ
N.V. KEMA,
Arnhem

K.A. NATER and J.H.N. VERHEUGEN
Reactor Centrum Nederland,
Petten N.H.

Vibratory compaction of fuel particles is a promising alternative to pelletized fuel in the production of fuel rods. VIBRASOL fuel rods, as developed in the Netherlands, use high density spherical fuel particles of distinct size fractions and narrow size distribution. These particles are produced through a sol-gel type process. With three particle fractions, smeared densities up to 92% T.D. have been obtained. Plutonium can be readily incorporated in the fuel. Pilot plant facilities for UO_2 and $(U,Pu)O_2$ are in operation to study the scaling up potential of this fabrication method. Irradiation of vibrasol fuel rods is in progress.

NUC. EN. MATUR., vol. 7, 1976

4. Results of QA Inspections at Fuel Vendors, *L. E. Tripp (NRC)*

In early 1973, the USAEC Directorate of Regulatory Operations (now USNRC, Office of Inspection and Enforcement) initiated a program of direct inspection of light-water reactor fuel vendors' (manufacturers) QA programs. Their QA programs were each subjected to a series of NRC team inspections utilizing the 18 quality assurance criteria of 10CFR Appendix B as the inspection standard.

Several deficiencies and weaknesses with respect to Appendix B criteria were identified with similar problems often identified at two or more vendors. These are listed in Refs. 1 and 2. Some of the most significant deficiencies or weaknesses are listed below:

1. There were inadequate quality assurance staffs.

2. Design engineers were not familiar with, or aware of, the design quality assurance requirements.

3. Design control procedures were incomplete.

4. Design analyses and calculations were not properly documented, checked, or identified.

5. Internal design control procedures were not being followed with respect to design checking and verification and certification of design computer programs.

6. Procurement documents did not require subvendors to have quality assurance programs consistent with the appropriate portions of Appendix B.

7. Sub-vendor deficiencies were not being resolved. Sub-vendor audits were ineffective or were not being performed per procedures.

8. Updated drawings were not at the appropriate manufacturing and inspection stations.

9. Fuel rod end-cap welding did not conform to requirements, qualification records were inadequate, and qualifications did not follow procedures.

10. There were no qualification requirements or documented qualifications of personnel performing special processes, such as welding and NDT testing.

11. Calibration programs were not being followed.

12. Nonconforming materials, parts, or components such as discrepant powder or tubing were not properly segregated from normal production items.

13. Internal audit procedures were inadequate.

14. Audits were not being conducted or were not in accordance with the prescribed frequency.

15. Audit findings were not properly transmitted to responsible management, resulting in poor or no follow-up corrective action.

The results of these inspections plus past fuel failure experience have shown there are certain critical quality assurance areas that need to be strengthened. These are summarized below.

1. The more serious fuel failure causes, in terms of amount of fuel affected, such as internal hydriding, densification and cladding collapse, and fuel-cladding mechanical interaction are at least as much design as fabrication related. This points out a serious fallacy in most fuel quality assurance programs, which tend to devote almost all quality assurance efforts to manufacturing activities. In the NRC inspections, design quality assurance programs were found, in general, to be significantly weaker than, or lagging the fabrication QA programs.

2. Utilization of inadequately qualified fabrication techniques (as well as designs) has also been a contributing factor to in-reactor fuel failures. The large number of deficiencies noted in the qualification of special processes and/or the personnel performing them show that qualification is still a major problem.

3. Adverse experience with sub-vendor-supplied components indicates a continuing need for the primary vendor (fuel fabricator) to exercise strong sub-vendor control. The large number of deficiencies noted under Criterion VII (Control of Purchased Material, Equipment, and Services) indicates that this is a weak area.

4. The number of deficiencies noted in audit programs (internal and external vendor audits) indicates that the optimum value of auditing as a part of the quality assurance system is not being achieved. This appears to be primarily a management problem.

5. Failure of in-house personnel to follow internal procedures continues to be a problem. This may be due to lack of knowledge of the existence of such procedures (indoctrination and training); procedures too numerous, lengthy, or complex; or lack of management emphasis on the implementation of procedures.

---

1. L. E. TRIPP, "Quality Assurance Programs for Nuclear Fuel Production in the United States," presented at IAEA Panel on Quality Assurance in Nuclear Fuel Manufacture, Vienna, Austria (Nov. 1974).

2. Inspection Reports, Dockets in 999000XX series (available in U.S. NRC Public Document Room, Washington, D.C.).

1. Quality Assurance in Nuclear Fuel Fabrication, *R. E. Trumble (Exxon Nucl. Co.)*

This paper presents (a) an evaluation of the status of quality assurance in nuclear fuel fabrication, (b) some of the generic things Exxon Nuclear Company has learned in developing and applying a quality assurance program, and (c) trends in the application of quality assurance to nuclear fuel fabrication.

The 18 criteria of Appendix B of 10CFR50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," have long been applied to nuclear fuel fabrication via the utilities, and more recently by the NRC directly. The 18 criteria represent an adequate approach to developing a systematic and disciplined method for controlling the design and fabrication of nuclear fuel.

However, broad criteria require interpretation. ANSI Standard N45.2 and daughter standards N45.2.1, N45.2., . . . undertake to amplify and interpret the requirements of Appendix B of 10CFR50 for nuclear power plants, and a standard, N45.2.21, is under preparation to interpret the requirements of Appendix B and N45.2 plus daughters for nuclear fuel fabrication. Nevertheless, each team of auditors seems to have its own distinct requirements for an acceptable Quality Assurance Program for nuclear fuel fabrication. This, plus the multiplicity of audits, imposes a significant burden on fuel fabricators.

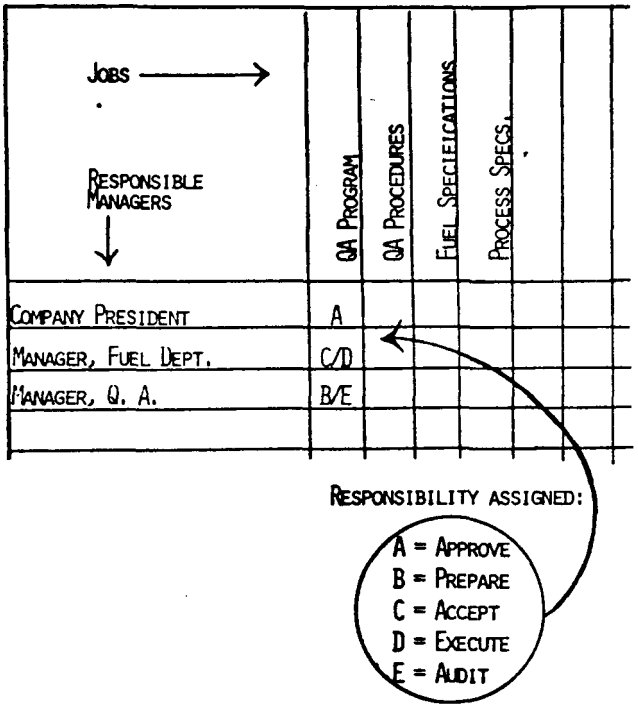


Fig. 1. Matrix of responsibility and accountability.

Thus, at the present time there is no doubt that nuclear fuel fabrication and their suppliers are subject to Appendix B requirements, but there does remain a question as to what this means. These areas of uncertainty lie usually in the fine structure of the program but do provide a happy hunting ground for auditors.

In developing and applying a quality assurance program and especially in imposing a quality assurance program on a supplier for the first time, we at Exxon Nuclear Company have identified six generic problems that can be avoided by foresight:

1. Quality must be built into a product; it cannot be "inspected-in." Building the quality begins with the top management of the company—those who create the value systems and evaluate and reward managers—and it extends down the line of command to the skilled craftsmen on the floor of the shop, and even to the janitor who sweeps the floor.

2. Quality cannot consistently be maintained for a complex product, with many people involved, without a formal quality assurance program.

3. A Quality Assurance Program is a means to an end, not an end in itself.

4. It may be a traumatic change to a vendor to be required, for the first time, to prepare written procedures for familiar processes and to get formal approvals; to keep detailed records; and to submit to external audits.

5. A Quality Assurance program may easily be made to be unnecessarily and unwisely restrictive.

6. A Quality Assurance program may fail to incorporate adequate methods for making changes in the program.

And finally, we have identified two trends, one generic and one that may be unique to the Exxon Nuclear Company.

The first trend is to extend the requirements of Appendix B to areas other than nuclear safety; and in the fuel fabrication area, reliability is the key word.

The second trend is to extend the principles of systematic and disciplined approach to providing a superior product embodied in Appendix B to running a business. One key element in this extension is to develop a matrix of Responsibility and Accountability covering essential tasks to be performed, and key managers to be assigned responsibility and accountability. An illustrative example of such a matrix is shown in Fig. 1.

ANS TRANS., vol. 22, 1975

15527 (IAEA-145, pp 487-492) NONDESTRUCTIVE QUALITY CONTROL METHODS USED DURING THE FABRICATION OF FUEL ELEMENTS. Watteau, M. (CEN, Saclay, France). 1972. (In French).

From meeting on NDT for reactor core components and pressure vessels; Vienna, Austria (29 Nov 1971).

The nondestructive quality control methods used during the fabrication of fuel elements with fritted ceramic fuel and metallic jacketing are described. These fuel elements usually consist of a rod of fuel pins maintained in a regular geometric arrangement by structural pieces. The nondestructive testing used during the fabrication of the pin include inspection of the pin constituents, of welds and of the corrosion behavior of the jacket, leak testing, determination of the distribution of the internal constituents, and inspection of the jacket seals. The quality control methods used during the fabrication of the structural pieces and on the completed fuel element are briefly reported. (JSR)

NSA, vol. 29, 1974

19715 (WCAP-4167-2, Encl.1) PLUTONIUM DIOXIDE-URANIUM DIOXIDE PELLET SPECIFICATION. (Westinghouse Electric Corp., Pittsburgh, Pa. Nuclear Fuel Div.). Mar 1971. 6p.

In EEI-Westinghouse plutonium recycle demonstration program progress report for the period ending December 1970.

The specification for fabrication of plutonium recycle pellet fuel for pressurized water reactors is presented. The fabrication requirements apply to solid cylindrical PuO_2-UO_2 fuel pellets in the sintered and ground condition and fabricated by mechanically mixing, cold pressing, and sintering. (H.D.R.)

NSA, vol. 26, 1972

11776 (WCAP-7800(Rev.2)) NUCLEAR FUEL DIVISION QUALITY ASSURANCE PROGRAM PLAN. (Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). Sep 1973. 41p.

Information on the Westinghouse Nuclear Fuel Division quality assurance program is presented under the following headings: organization and responsibilities; design control; procurement document control; instructions, procedures, and drawings; document control; control of purchased material, equipment, and services; identification and control of materials, parts, and components; control of special processes; inspection; test control; control of measuring and test equipment; handling, storage, and shipping; inspection, test, and operating status; nonconforming materials, parts or components; corrective action; quality control records; and audits. (JWR)

NSA, vol. 29, 1974

13065 (WCAP-7800(Rev.4-A)) NUCLEAR FUEL DIVISION QUALITY ASSURANCE PROGRAM PLAN. (Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). Mar 1975. vp. Westinghouse Electric Corp., Pittsburgh, PA.

The Westinghouse quality assurance program for the design, procurement, and manufacturing activities for nuclear fuel assemblies and associated reactor core components is presented. (JWR)

NSA, vol. 32, 1975

11096 (NP-20419) QUALIFICATION OF WESTINGHOUSE POWDER SOURCES. (Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). May 1975. 25p. Westinghouse Electric Corp., Pittsburgh, PA.

A basis for qualifying fuel with respect to the Westinghouse densification model is presented. The qualification basis is founded on the observations that relative stability of fuels increases with increasing mean grain size and that stability of pores decreases with their diameter. In addition to the grain size and fine porosity criteria, the thermal resintering behavior of Westinghouse fuel is explored. Westinghouse data indicates that for fuel with certain grain sizes most densification is the result of removal of pores. These observations have been utilized to select limits of grain size and volume of fine porosity and thermal resintering which assure the adequacy of new process fuel. The controls on this process as described in WCAP-8218-P-A provide additional assurance of the continued acceptability of the fuels once the process is qualified. (auth)

NSA, vol. 32, 1975

SUMMARIES OF RECENT MEETINGS

IAEA International Seminar on Nuclear Fuel Quality Assurance
(Oslo, Norway, 24-28 May 1976)

The objectives of the seminar were to provide educational lectures on the basic concepts of Quality Assurance (QA) and Quality Control (QC) as applied to nuclear fuels and to review the current methods of applying these in practice. These goals were achieved.

On the basis of the presentations and discussions at the meeting recommendations for future Agency activities in this area have been prepared. These include expansion of present efforts, regional seminars in the developing countries, a central information service in QA and QC development activities, and an Advisory Group on HTR fuel QA development.

The widespread interest in the topic of QA for nuclear fuels was demonstrated by the presence of participants from 25 countries. Furthermore, there was very active discussion in all of the sessions, especially in response to the lead lectures with a basic educational purpose. The participants represented a remarkable range of interests including fuel manufacturers, users (utilities and their consultants), regulatory bodies, process development personnel, and technicians from developing countries, while the discussions covered topics from the general philosophy of QA to details of QC related to LWR, LWR recycle, FBR, and HTR fuels.

The need for more detailed regional seminars on this topic, aimed at assisting the developing countries, was emphasized by the presence (and discussion period questioning) of 16 participants from nine developing countries.

To summarize, the most important points raised during the seminar discussion periods were: -

— Vendor-utility interaction — the question of how much information should be provided by the fuel manufacturer (the vendor) to the utility which is purchasing fuel, and in what form and how early this information should be provided, was a recurring subject of discussion. Opinion was divided on these points with some participants happy with their present arrangements and others not. It appears that the information transfer is basically set out in the individual vendor-utility contracts — with some indirect influence from the regulatory bodies through their requirements on the utility. Any conflict that exists appears to come from those utilities and their QA consultants which desire more extensive control over the product they are buying. There is a natural reluctance on the part of vendors to permit this but there seems to be an increasing trend toward release of more information thus involving both parties in closer co-operation. The discussion and illumination of these problems fulfilled one of the objectives of this seminar.

  The close co-operation of vendor and utility can only benefit all parties — in particular since the utility has the best source of the real fuel performance data needed for feed-back to fuel design and QA evaluation.

(Cont'd pg 139)

- QA of fuel design — this area of fuel QA recurringly appeared in the discussions. There was general agreement that it was the weakest point in the QA scheme at present and efforts are needed to remedy this problem. No general statement of how to accomplish this was made.
There are apparently two basic difficulties which confound attempts to move rapidly in design QA. One is the strong commercial significance of improved design and the other is the fact that much of the data on fuel failure is in the control of the utilities who have generally not been prepared for in-depth fuel examinations of a diagnostic nature. Nevertheless, attempts are under way to pool the utilities and vendors' information into data banks and connect fuel performance with design and specifications. It is felt these efforts were encouraged and the significance of the problem highlighted through this seminar.

- Fuel Specifications — this topic was approached from two viewpoints. There was an obvious interest on the part of fuel purchasers (developing countries and utilities) in getting as much detail of the fuel specifications as possible. This information is, however, not normally made available except through individual buyer-seller contracts.

  On the other end of the scale, there were comments by the manufacturers that often specifications were set without proper regard for the latest technical information on fuel performance and for the realities of manufacturing processes and testing capabilities. This problem will be resolved when the fuel design activities are properly meshed with the full QA system.

- Fuel Failure and QA — the costs of fuel failures and the desire for increased safety were mentioned as strong justifications for fuel QA. programs. One author quoted the cost of each fuel failure as being US$45,000.- and another cited the cost for a one-day outage of a 500 MWe plant as US$100,000. However, these figures should be applied to fuel with care. Advantage is usually taken of any reactor shut down to carry out several tasks, thus reducing the cost to be assigned to any one cause. Further, the value of the quality assurance program in reducing fuel failures is difficult to separate from the design improvements resulting from R&D programs initiated outside the QA program. It should also be recognized that fuel failures may also result from improper reactor operation not related to the fuel QA.

"Meetings on atomic energy", vol. 8, no. 4, 1976

Blank Page

V. ABSORBER OR BURNABLE POISON ROD FABRICATION

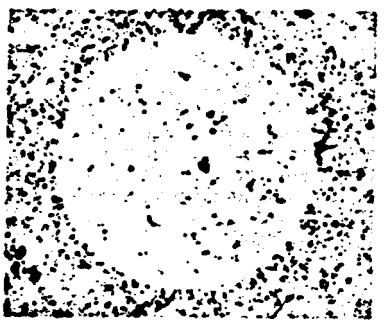
Blank Page

3,759,786

SINTERED NUCLEAR FUEL CONTAINING MOLYBDENUM COATED GADOLINIUM OXIDE MICROSPHERES

Giancarlo Abate-Daga and Ignazio Amato, Turin, Italy, assignors to Fiat Societa per Azioni, Turin, Italy

Filed Dec. 16, 1969, Ser. No. 885,524
Claims priority, application Italy, Feb. 25, 1969, 50,722/69
Int. Cl. C09k 3/00; G21c 3/20
U.S. Cl. 176—69 2 Claims



Sintered nuclear fuel, uranium dioxide in particular, containing a consumable poison of rare-earth oxide or metal boride uniformly distributed in the form of coated microspheres of 10 to 20,000 microns diameter.

U.S. PATENTS, 1973

11368 EXPERIMENTAL AND THEORETICAL INVESTIGATIONS ON SOLID BURNABLE POISON BURNUP OF MODEL ARRANGEMENTS. Ahlf, J.; Anders, D.; Greim, L.; Knoth, J.; Kolb, M.; Mittelstaedt, B.; Mueller, A.; Schwenke, H. (Gesellschaft fuer Kernenergieverwertung in Schiffbau und Schiffahrt m.b.H., Geesthacht-Tesperhude (F. R. Germany)). Atomkernenergie; 25: No. 2, 91-97(1975). (In German)

From KTG-special meeting "long-term behavior"; Stuttgart, F. R. Germany (25 Feb 1975). 13 figs.; 2 tabs.; 8 refs.

Two experiments are reported to improve the methods for computation and measurement of burnable poison rod burnup. In the first experiment, two nine-rod bundles in a 3 × 3 arrangement are irradiated during several irradiation periods in the research reactor Geesthacht. Each bundle consists of eight outer rods containing fuel and one inner rod containing poison (^{10}B or ^{113}Cd). The burnup of the fuel and the burnable poison is measured by non-destructive methods after each irradiation period and then compared with results of a burnup calculation. In the second experiment, two poison rods with different cadmium concentrations and one rod containing boron are irradiated during several irradiation periods in the research reactor Geesthacht. The burnup is determined after each irradiation period by reactivity measurements and its result compared to computed effective absorption cross-sections of the rods by aid of a calibration curve. For both experiments, the experimental and theoretical results for the poison burnup are found to be within the error limits of the measurements. (auth)

NSA, vol. 33, 1976

1740 (GKSS-72-E-21) TEST FUEL ELEMENTS CONTAINING SOLID BURNABLE POISON FOR BURNUP INVESTIGATIONS. Ahlf, J.; Greim, L.; Knoth, J.; Kolb, M.; Schwenke, H. (Gesellschaft fuer Kernenergieverwertung in Schiffbau und Schiffahrt m.b.H., Hamburg (West Germany)). 1972. 20p. (CONF-721138-8). Dep. NTIS (U. S. Sales Only) $3.00.

From symposium on irradiation facilities for research reactors; Teheran, Iran (6 Nov 1972).

Data are evaluated for testing and confirming theoretical models and codes for burnup calculations on reactor cores containing solid burnable poisons for excess reactivity compensation. Test fuel elements consisting of a square array of 3 × 3 rods, the outer eight containing fuel, the inner one containing the poison (boron or cadmium), are irradiated in reflector positions of FRG-2. At intervals of about six months the irradiation is interrupted for burnup measurements. The reactivity worth of the whole test element is measured in a core of fresh MTR elements relative to unirradiated reference inserts by means of calibrated control rods. The burnup of the fuel rods and its axial distribution is measured by γ-scanning. For this purpose a scanning machine with collimator and Ge(Li)-γ-spectrometer is installed in a hot cell. The poison burnup is determined by transmission measurements with monochromatic

NSA, vol. 28, 1973

30426 INVESTIGATIONS ON THE BURN-UP OF BURNABLE POISON RODS CONTAINING Cd. Ahlf, J.; Kolb, M.; Krueger, A.; Lobmeyr, M.; Mueller, A. (Gesellschaft fuer Kernenergieverwertung in Schiffbau und Schiffahrt mbH, Geesthacht, Ger.). Atomkernenergie; 19: No. 2, 101-6(Apr 1972). (In German).

In order to test the codes used at Geesthacht to calculate the burn-up of burnable poison rods, two Cd–Mg rods (1 cm diam. × 20 cm) were irradiated stepwise in a mean thermal neutron flux density of 5 to 6 × 10^{13} cm^{-2} s^{-1} until complete burn-up of ^{113}Cd. The neutron fluence for the irradiation steps was found by means of Co–Al activation wires. The reactivity reached by each rod was obtained within a small water-moderated subcritical assembly. The burn-up process measured in this way coincides with the calculated one which shows a distinct difference between "surface burn-up" for the high Cd density rod (70.4 weight percent Cd) and the "volume burn-up" of the lower Cd density rod (33.3 weight percent Cd.). (12 references) (auth)

NSA, vol. 26, 1972

22790 METHOD FOR PRODUCING POLYCRYSTALLINE BORON NITRIDE. Alexeevskii, V. P.; Bochko, A. V.; Dzhamarov, S. S.; Karpinos, D. M.; Karyuk, G. G.; Kolomiets, I. P.; Kurdyumov, A. V.; Pivovarov, M. S.; Frantsevich, I. N.; Yarosh, V. V. US Patent 3,876,751. 8 Apr 1975. Priority date 18 Dec 1972, Ukrainian SSR. 6p.

A mixture containing less than 50 percent of graphite-like boron nitride treated by a shock wave and highly defective wurtzite-like boron nitride obtained by a shock-wave method is compressed and heated at pressure and temperature values corresponding to the region of the phase diagram for boron nitride defined by the graphite-like compact modifications of boron nitride equilibrium line and the cubic wurtzite-like boron nitride equilibrium line. The resulting crystals of boron nitride exhibit a structure of wurtzite-like boron nitride or of both wurtzite-like and cubic boron nitride. The resulting material exhibits higher plasticity as compared with polycrystalline cubic boron nitride. Tools made of this compact polycrystalline material have a longer service life under impact loads in machining hardened steel and chilled iron. (Official Gazette)

NSA, vol. 32, 1975

32120 (TID-16597) APPLICATIONS OF ABSORBER MATERIALS IN NUCLEAR REACTOR TECHNOLOGY. W. Kermit Anderson (Knolls Atomic Power Lab., Schenectady, N. Y.). [Sept. 4, 1962]. Contract [W-31-109-eng-52]. 15p.

The neutron absorption (high cross section) materials used for shutting down reactors are discussed with respect to selection and properties. The selection is based on abundance, properties, fabrication technology, and economic aspects. Materials discussed include boron, cadmium, lanthanons, lithium-6, boron-10, cadmium-indium-silver alloys, hafnium, and europium oxide dispersions. (N.W.R.)

NSA, vol. 16, 1962

23288 NEUTRON ABSORBER MATERIALS FOR REACTOR CONTROL. W. Kermit Anderson and J. S. Theilacker, eds. (Division of Reactor Development, AEC). 1962. 878p. $3.00(GPO).

Information on boron, silver alloys, and rare earth compounds along with a summary of information on hafnium obtained in connection with the development of control rods and burnable poison elements is presented. The physical, chemical, and mechanical requirements for control rods and burnable poison elements which influence selection and development of materials with certain properties are covered. The components made from these control materials, how they are fabricated, and how they are used in nuclear reactors are described. (M.C.G.)

NSA, vol. 16, 1962

31406 (BAW-3647-27) PHYSICS VERIFICATION PROGRAM PART III, TASK 9. Quarterly Technical Report, April–June 1973. Baldwin, M. N.; Webb, H. W. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Research and Development Div.). Dec 1973. Contract AT(11-1)-3109. 20p. Dep. NTIS $3.00.

An experimental program is in progress to measure the fission rate in UO_2 fuel at selected positions within a critical assembly and to compare it with the neutron absorption rate in the boron of a simulated lumped burnable poison material. (11 references) (auth)

NSA, vol. 29, 1974

10251 (BAW-3647-29) PHYSICS VERIFICATION PROGRAM PART III, TASKS 9 AND 10. Quarterly Technical Report, October–December 1973. Baldwin, M. N.; Warren, H. D.; Webb, H. W. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Research and Development Div.). Jul 1974). Contract AT(11-1)-3109. 53p. Dep. NTIS $5.75.

Experimental Task 9 was conducted to compare the fission rate within the fuel rods of a critical assembly with the neutron absorption rate in adjacent pins containing simulated lumped burnable poison (LBP) material. The experimental assembly, a water-moderated, slightly enriched UO_2 lattice perturbed by the simulated LBP and dissolved boric acid, is arranged to mock up the fuel elements of a commercial pressurized water reactor (PWR). These results will serve to check analytical techniques used in the design of large PWRs. Experimental Task 10 was conducted to obtain data for use in assessing the relative magnitude of the neutron and gamma induced signal components from an all-solid, prompt-responding incore detector. The measurements were conducted in three separate, slightly enriched UO_2 lattices which are designed to provide a wide range for the neutron-to-gamma ratio and for the neutron spectrum. (auth)

NSA, vol. 31, 1975

10252 (BAW-3647-30) PHYSICS VERIFICATION PROGRAM PART III, TASK 11. Quarterly Technical Report, January–March 1974. Baldwin, M. N. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Research and Development Div.). Jul 1974. Contract AT(11-1)-3109. 19p. Dep. NTIS $4.00.

The reactivity and power perturbation effects of poison pins in mockups of 17 by 17 fuel assemblies were measured. The experimental facility—a water-moderated, slightly enriched UO_2 lattice—is arranged to mock up the fuel assemblies of a commercial pressurized water reactor. Each assembly contains 24 poison pin positions. (auth)

NSA, vol. 31, 1975

27956 (BAW-3647-31) PHYSICS VERIFICATION PROGRAM. Part III. Task II. Quarterly Technical Report, April–June 1974. Baldwin, M. N. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Research and Development Div.). Nov 1974. Contract AT(11-1)-3109. 18p. Dep. NTIS $4.00.

In experimental Task 11, the reactivity and power perturbation effects of poison pins in mockups of 17 by 17 fuel assemblies are being measured. The experimental facility—a water-moderated, slightly enriched UO_2 lattice—is arranged to mock-up the fuel assemblies of a commercial pressurized water reactor. Each assembly contain 24 poison pin positions. (12 references) (auth)

NSA, vol. 31, 1975

24604 (BAW-3647-32) PHYSICS VERIFICATION PROGRAM, PART III, TASK 11. Quarterly Technical Report, July–September 1974. Baldwin, M. N. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Research and Development Div.). Feb 1975. Contract AT(11-1)-3109. 15p. Dep. NTIS $4.00.

In experimental Task 11, the reactivity and power perturbation effects of poison pins in mockups of 17 by 17 fuel assemblies are being measured. The experimental facility–a water-moderated, slightly enriched UO_2 lattice–is arranged to mock up the fuel assemblies of a commercial pressurized water reactor. Each assembly contains 24 poison pin positions. (auth)

NSA, vol. 31, 1975

28541 (HEDL-TME-71-191) EVALUATION OF CARBIDE VENDOR PREQUALIFICATION PELLETS. Final Report. Barr, M. J. (Hanford Engineering Development Lab., Richland, Wash.). Nov 1971. Contract AT(45-1)-2170. 44p. Dep. NTIS.

A boron carbide vendor prequalification program was undertaken to evaluate the capability of industry to produce boron carbide pellets to a provisional FFTF specification and to provide a technical base from which a formal boron carbide pellet specification could be written. Three potential vendors' products were evaluated with none of the three meeting every requirement of the provisional specification. Problem areas or deficiencies are identified and discussed. Data obtained from this evaluation have been incorporated into RDT Standard E 6-30T, Absorber Pin Boron Carbide Pellet. (auth)

NSA, vol. 26, 1972

45952 DYSPROSIUM BOROCARBIDES. Bauer, Josef; Debuigne, Jean (Institut National des Sciences Appliquees, Rennes, France). Compt. Rend., Ser. C; 274: No. 13, 1271-4(27 Mar 1972). (In French).

A ternary Dy–B–C system was studied using x-ray diffraction and metallography on samples quenched after melting. The existence of the DyB_2C_2, DyB_2C, DyBC and Dy_2BC_2 phases was verified. The structure of the three first phases and that of the YB_2C_2, YB_2C, and YBC phases are isotypic. A unit cell is proposed for Dy_2BC_2. (France)

NSA, vol. 26, 1972

37816 (CONF-720420-5) BOREON CONCENTRATION GRADIENT FOR IMPROVED THERMAL REACTOR PERFORMANCE OF BORON-STAINLESS STEEL CONTROL RODS. Beaver, R. J.; Richt, A. E. (Oak Ridge National Lab., Tenn.). [1964]. 29p. Dep. NTIS.

From seventeenth annual ISA analysis instrumentation symposium; Houston, Tex. (19 Apr 1971).

The performance of conventional 2 to 3 wt % ^{10}B-stainless steel alloys and powder dispersions is limited in thermal reactors because the $^{10}B(n,\alpha)$ reactions are localized at and near the surface of the material. Significant swelling occurs after a relatively short exposure, and additional exposure creates drastic damage to the microstructure in the localized regions. By designing a symmetrical boron concentration gradient in which the ^{10}B is limited at the surface and increases in a prescribed manner to higher concentrations below the surface, the performance capability of boron-stainless steel neutron absorbers can be increased substantially. Test results for boron concentration gradient rods are presented. 7 references. (auth)

NSA, vol. 26, 1972

16514 (ORNL-4368) STATUS OF EUROPIUM COMPOUNDS FOR REACTIVITY CONTROL IN NUCLEAR REACTORS. Beaver, R. J.; Martin, M. M. (Oak Ridge National Lab., Tenn.). Feb. 1969. Contract W-7405-eng-26. 17p. (CONF-680324-1). Dep. CFSTI.

From Materials Engineering and Sciences Conference and Exposition, Philadelphia, Pa.

Europium is an attractive element for controlling excess reactivity in thermal reactors because of its high thermal neutron absorption cross section of 3980 barns and its inherent resistance to irradiation damage. Compounds of europium, particularly europium sesquioxide (Eu_2O_3), europium titanate ($Eu_2Ti_2O_7$), and europium molybdate ($Eu_{5.3}MoO_{11}$), have been developed for these applications. Procedures are described for preparing these compounds and dispersing them in stainless steel and aluminum. Chemical stability, physical properties, and irradiation behavior of these dispersions are reviewed. (auth)

NSA, vol. 23, 1969

49999 BORON CONCENTRATION GRADIENT FOR IMPROVED THERMAL REACTOR PERFORMANCE OF BORON-STAINLESS-STEEL CONTROL RODS. Beaver, R. J.; Richt, A. E. (Oak Ridge National Lab., Tenn.). Nucl. Technol.; 16: No. 1, 187-96(Oct 1972).

The performance of conventional 2 to 3 wt % ^{10}B-stainless steel alloys and powder dispersions is limited in thermal reactors because the $^{10}B(n,\alpha)$ reactions are localized at and near the surface of the material. Significant swelling occurs after a relatively short exposure, and additional exposure creates drastic damage to the microstructure in the localized regions. By designing a symmetrical boron concentration gradient in which the ^{10}B is limited at the surface and increases in a prescribed manner to higher concentrations below the surface, the performance capability of boron-stainless steel neutron absorbers can be increased substantially. This concept was exploited in the 10 MW SM-1 Reactor. The Neutron Absorber Section Assembly of the Control Rod is a rectangular parallelepiped consisting of four 0.090-in.-thick plates clad with 0.032-in.-thick austenitic stainless steel. The ^{10}B concentration at the surface of the plate was established as 1 wt % ^{10}B, and in a programmed manner increased to 3 wt % ^{10}B 0.024 in. below the surface. For an intended fluence of 4.3×10^{20} nvt (thermal) this technique limits the ^{10}B consumption in any volume increment to 36×10^{20} atoms/cm^3 (maximum). Thus, the allowable average ^{10}B burnup for the onset of swelling can be increased from 8 at. % burnup, the limit for conventional dispersions, to 20 at. % burnup. This improvement was demonstrated by the post-irradiation examination of a boron-gradient neutron absorber which contained ^{10}B dispersed in Type 200 austenitic stainless steel, and operated for six years in the central region of the active lattice of the SM-1 Reactor. (7 references) (auth)

NSA, vol. 26, 1972

15562 (BDX-613-227(Rev.)) BORON SHOT FABRICATION. Final Report. (Bendix Corp., Kansas City, Mo.). Nov 1971. Contract AT(29-1)-613. 78p. Dep. NTIS.

Boron shot of 0.015 ± 0.005 in. diameter was produced in up to 5 g quantities by a shot tower method. Ten- to 100-g charges of molten boron at 3,800 to 4,200°F were pneumatically squirted through 0.007 to 0.015 in. diameter orifices. Problem areas include material compatibility, initiation of streaming, and shot quenching. (auth)

NSA, vol. 26, 1972

49996 (RD/B/N-2274) BURNABLE POISONS: AN EXPERIMENTAL DEPLETION AND COMPARISON WITH THEORETICAL PREDICTION. Beynon, A. J.; Gavan, T. (Central Electricity Generating Board, Berkeley (England). Berkeley Nuclear Labs.). Mar 1972. 40p. Dep. NTIS (U. S. Sales Only).

The accuracy of theoretical methods of predicting the self-shielded depletion of a cylinder of a gadolinium-based burnable poison was determined by irradiating a column of discs of magnesia-diluted gadolinia in the reflector of DIDO. The flux varied along the column by a factor of 10 and depletions on discharge varied from 90% at the bottom to 10% at the top. Neutron exposure as a function of time was determined by regular flux measurements along the column, and after irradiation radiographs were taken. Discs were then selected for isotopic analysis and theoretical methods were used to determine the residue of poison expected in each. Values of poison residue determined by a theoretical treatment in which the flux was isotropic and azimuthally symmetric showed a peak deviation from measured values of 1.0 ± 1.4% of the initial value. In a treatment where the point incident vector flux was of the form $\cos\theta\, d\Omega$, where θ was the angle between the direction of $d\theta$ and the horizontal the depletion was overestimated and deviations ranged up to 4.5 ± 1.7% of the initial value with a tendency to increase with depletion. (UK)

NSA, vol. 26, 1972

42635 SOME HIGHLIGHTS OF THE THREE YEAR BELGIAN CRITICAL EXPERIMENT AND CODE CALIBRATION WORK ON UO_2-PuO_2-H_2O LATTICES. Bindler, L. (Societe Belge Nucleaire pour l'Industrie Nucleaire, Brussels); Charlier, A.; Debrue, J.; Leenders, L.; Van den Broeck, H. pp 178-81 of Reaktortagung, Bonn, 1971. Bonn; Deutsches Atomforum E. V. (1971).

From Reactor meeting; Bonn, Germany (30 Mar 1971). See CONF-710346.

A comment is given on the basic critical experiment program. Moreover such problems are investigated by experiment and calculation as arise from the application of plutonium fuel. The most important problems which are investigated are the power distribution (for example: plutonium fuel assembly in Dodewaard) and the effects on the reactivity in the case of boron poisoning of the water moderator. (INIS)

NSA, vol. 26, 1972

41006 SEPARATION OF BORON ISOTOPES. Blum, J. M. (Societe Isobore, Sevres, France); Marteau, S. Energ. Nucl. (Paris); 14: No. 1, 33-7(1972). (In French).

The various ways of separating B isotopes are reviewed. The utilization of an ion exchange process is discussed in detail. The plant of the Isobore company is described. The various uses of the different B isotopes are also described in detail. (France)

NSA, vol. 26, 1972

30191 DEVICE FOR CONTROLLING THE REACTIVITY OF NUCLEAR REACTORS. Braun, Wolfgang (to Siemens Aktiengesellschaft). U. S. Patent 3,652,394. 28 Mar 1972. Priority date 23 May 1969, Germany.

A device is described for controlling the reactivity of a nuclear reactor cooled by light water. The reactor core is formed of fuel elements, and a tube system passing through the reactor core. The tube system is transversed by a water-soluble neutron absorber of adjustable concentration. The system includes a plurality of substantially vertically extending fingerlike tubes always fully inserted in the fuel elements; a spiderlike holder holding the fingerlike tubes together at the upper ends serves as distributor for the neutron absorber to the fingerlike tubes. A holder rod extending from the spiderlike distributor serves as a central supply line for the neutron absorber to the spiderlike distributor. (auth)

NSA, vol. 26, 1972

32287 NUCLEAR POISON. (to Centre d'Etude de l'Energie Nucleaire). Netherlands Patent 7,206,360. 10 May 1972. 5p. (In Dutch).

The invention provides a burnable poison for thermal nuclear reactors. The poison dysprosiumaluminate (3 Dy_2 V_1 5 Al_2 0_3) is added as particles to the fuel during the fabrication process. (NL)

NSA, vol. 31, 1975

3,849,248

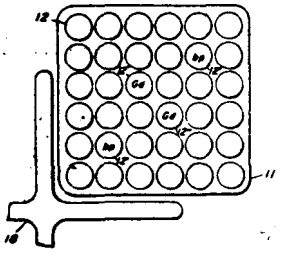
SAMARIUM COMPENSATION FOR NUCLEAR REACTOR FUEL

Frederick R. Channon, and David L. Fischer, both of San Jose, Calif., assignors to General Electric Company, San Jose, Calif.

Continuation of Ser. No. 799,467, Feb. 14, 1969, abandoned. This application Oct. 27, 1971, Ser. No. 193,221

Int. Cl. G21c *3/04, 7/02, 7/24*

U.S. Cl. 176—68 **9 Claims**



1. A fuel assembly for use with other fuel assemblies in a thermal nuclear reactor core, said core including control rods which are selectively movable into and out of said core to control reactivity thereof, said fuel assembly comprising an array of a plurality of spaced fuel elements containing new nuclear fuel initially lacking an equilibrium amount of samarium, said new fuel providing an excess of reactivity to compensate for said equilibrium amount of samarium which said new fuel will contain after a period of operation in said core, and an initial amount of dilute fixed gadolinium burnable poison in said fuel assembly to absorb neutrons during the period of samarium build-up in said new fuel to equilibrium to compensate for said initial lack of samarium and the said initial excess of reactivity whereby said burnable poison substantially depletes out of said fuel as the samarium builds up to the equilibrium amount, said dilute gadolinium poison having a density and distribution in the fuel assembly such that the poison will not be substantially self-shielding, said gadolinium being particularly useful in said fuel assembly as compared to other burnable poison in that it has both a neutron capture cross section which decreases more rapidly than 1/v and than the cross section of samarium-149 over thermal neutron energy range of from 0.005–0.1 electron volts, and also provides either an increased "cold" shutdown margin or an increased "hot" operating reactivity margin the initial amount of said burnable poison being substantially defined by:

burnable poison atoms/fuel atoms = $Y\,\sigma_f'/\sigma_c^{bp}$

where

Y is the fractional yield of Sm-149 atoms per fission of fuel,

σ_f' is the microscopic neutron fission cross section of the fuel, and

σ_c^{bp} is the microscopic neutron capture cross section of the burnable poison.

U.S. PATENTS, 1974

15892

Ag–In–Cd COULD REPLACE Hf FOR PWR CONTROL RODS. I. Cohen, E. F. Losco, and J. D. Eichenberg (Westinghouse Electric Corp., Pittsburgh). Nucleonics 16, No. 8, 122-7(1958) Aug.

The results of tests made on a silver–15 wt. % indium–5 wt. % cadmium alloy indicate that it could replace hafnium in control rods of future pressurized-water reactor cores. The effectiveness of the alloy for neutron capture is 99% that of Hf, and the decrease is small over the core life. It has good corrosion resistance. The alloy has sufficient yield strength to resist scramming loads of PWR control rods. There is no change in appearance and mechanical strength improves with irradiation. The melting, forging, extrusion, and machining are readily performed. The alloy would be preferable to hafnium because of greater availability and lower cost. (J.S.R.)

NSA, vol. 12, 1958

265235 Anon. Standard specification for nuclear-grade boron carbide powder. *Annu. Book ASTM Stand.* (1975). (no.45) p. 342–343. ASTM-C–750–74.

Standards are specified for the basis of purchase and quality of product of BC powders ranging from fine to coarse powders. The quality control standards are set so as to assure that the BC powders meeting these specifications are suitable for nuclear applications. Chemical requirements of the powders are tabulated.

BORON CARBIDES chemical analysis specifications

Atomindex, vol. 7, 1976

10672
METALLURGICAL DESIGN AND PROPERTIES OF SILVER–INDIUM–CADMIUM ALLOYS FOR PWR CONTROL RODS. Presented at Nuclear Engineering and Science Conference, held at Chicago, March 17 to 21, 1958. Preprint 19, Session 8. I. Cohen, E. F. Losco, and J. D. Eichenberg (Westinghouse Electric Corp., Pittsburgh). New York, American Institute of Chemical Engineers, 1958. 38p.

The metallurgical design and properties of recently developed silver–indium–cadmium alloys are discussed. These alloys show considerable promise of fulfilling metallurgy, physics, and mechanical design requirements of control rods for PWR (Pressurized Water Reactor). By extension of the Hume-Rothery electron concentration rule, ternary and quaternary Ag-base alloys were designed for metallurgical stability and their properties determined. Of particular interest is a Ag–15 wt.% In–5 wt.% Cd alloy which is comparable to hafnium in neutron absorption characteristics. The alloy, a single-phase solid solution, is designed to be dimensionally stable under prolonged exposure at reactor operating conditions. Its corrosion resistance in pressurized water of varying chemistry was explored in the temperature range of 500 to 650°F and found to be remarkably high, thus permitting its operation without protective cladding. Physical and mechanical properties are described. Yield strength is lower than that of hafnium but entirely adequate for industrial power reactors. The alloy is quite ductile and can be easily fabricated by usual forging, rolling, and extruding techniques. Properties of full-size Ag–15 wt.% In–5 wt.% Cd PWR-type control rods are discussed. (auth)

NSA, vol. 12, 1958

48334 IRRADIATION BEHAVIOR OF BORON CARBIDE. Copeland, G. L.; Donnelly, R. G.; Martin, W. R. (Oak Ridge National Lab., Tenn.). Nucl. Technol.; 16: No. 1, 226-37 (Oct 1972).

Boron carbide is the prime candidate material for neutron absorbers in fast breeder reactors and the Fast Test Reactor. Important data required for design of control rods for these reactors concern swelling, gas release, and structural integrity of boron carbide under the expected operating conditions. Data for irradiations of boron carbide powders in a thermal reactor and powders and pellets in a fast reactor are presented and discussed in terms of expected performance in a fast reactor. The most important variable in determining irradiation behavior appears to be the mobility of the helium produced. The helium is not trapped in the lattice but precipitates. Denuded zones and absence of bubbles on grain boundaries suggest that helium diffuses rapidly along the grain boundaries. This is a major source of gas release. (auth)

NSA, vol. 26, 1972

8. Transmission Electron Microscopy of Irradiated Boron Carbide,* *G. L. Copeland, C. K. H. DuBose, D. N. Braski (ORNL)*

Boron carbide has been used extensively as a neutron absorber material in various types of nuclear reactors. However, transmission electron microscopy, one of the more useful tools for studying irradiation damage, has been unavailable for this material because of the inability to produce thin sections. Boron carbide has resisted successful thinning by conventional techniques due to its hardness, brittleness, and chemical inertness, and because some voids are generally present that are attacked preferentially as the film is thinned. These difficulties have been overcome through use of an ion bombardment micromilling machine.

*Sponsor: J. E. Cunningham.

Special care is necessary to prevent breakage during mechanical thinning prior to ion bombardment thinning. A high density of microcracks in the irradiated materials resulted in severe breakage during the mechanical thinning. We were able to examine high-density boron carbide by preparing thin foils (~ 0.003 in. thick), irradiating the foils in a capsule, and loading the foils directly into the ion micromilling machine for thinning with no mechanical thinning after irradiation. We have examined both irradiated and unirradiated boron carbide powders by dispersing the powders in an aluminum matrix by vacuum-hot-pressing.[1] The ion milling machine thins the aluminum and boron carbide at the same rate so that uniform thin films may be obtained.

The voids shown in the unirradiated pellet in Fig. 1a are produced in the Kirkendall effects, as excess boron in the center of the particle diffuses to the edge of particle during hot pressing.[2] Some grown-in dislocations and a large amount of twinning are typical of the unirradiated structure. Irradiation at 350 to 400°C for up to six weeks does not affect these features. Figure 1b shows a twinned region in an irradiated sample. The black spots indicative of a large amount of lattice strain are uniform over the entire field, but are out of contrast in some of the differently oriented regions. This photograph is from a thin disk (0.003 in. thick by ~$\frac{1}{8}$ in. in diameter) which was irradiated in a capsule in the Oak Ridge Research Reactor (ORR) for three weeks (fluence 1.8×10^{20} n/cm^2) at about 400°C. The sample showed a large amount of microcracking and was extremely brittle. Similar damage is observed in boron carbide particles irradiated in ORR capsules at about 350°C to 8% burnup of $^{10}B(16 \times 10^{20}$ reactions/cm^3). The high irradiation damage density is uniform throughout the samples with no observable change near grain boundaries, twin boundaries, intragranular voids, or particle edges. No helium bubbles or other evidence of reaction product agglomeration was resolvable.

1. C. K. H. DUBOIS, D. N. BRASKI, and G. L. COPELAND, "Fuels and Materials Development Program Quart. Progr. Rept. September 30, 1970," ORNL-4630, pp. 168-71, Oak Ridge National Laboratory.

2. G. L. COPELAND, R. S. MATEER, and W. R. MARTIN, "Characterization of Boron Carbide for Fast Reactor Absorbers," *Trans. Am. Nucl. Soc.*, 13, 100 (1970).

3. S. THOMPSON, "Fuel Shuffle Analysis–145 Fuel Assembly Core," NPGD-TM-173, Babcock & Wilcox (February 1972).

ANS TRANS., vol. 14, 1971

49414 (RT/MET-(70)4) FABRICATION OF CONTROL RODS. Dalmastri, B.; Fizzotti, C.; Lo Tenero, B. (Comitato Nazionale per l'Energia Nucleare, Rome (Italy)). 21 Mar 1970. 24p. (In Italian). Dep. NTIS (U. S. Sales Only).

Techniques employed for the fabrication of control rods were investigated and discussed. Two types of control rods were fabricated: metallic (Ag–In–Cd) rods and ceramic (B_4C) rods. Metallic rods were prepared by casting, swaging, and canning in stainless steel. Ceramic rods were prepared by vibrational compacting the B_4C powder in stainless steel cans. (auth)

NSA, vol. 25, 1971

42822 CONTROL ROD FABRICATION EXPERIMENTS. Dalmastri, B.; Fizzotti, C.; Lo Tenero, B. (CNEN, Rome). Com. Naz. Energ. Nucl., Notiz.; 16: No. 12, 61-7(Dec 1970). (In Italian).

Techniques were developed for the fabrication of two types of control rods: boron carbide rods and Ag–In–Cd rods. The techniques were described. (J.S.R.)

NSA, vol. 25, 1971

27434 (CONF-701074-P2, pp 577-593) RADIATION STABILITY AND EFFECTIVENESS OF DYSPROSIUM BASED ABSORBING MATERIALS. Emelyanov, I. Ya.; Grebennikov, R. V.; Tabachenkov, L. F. 1970. Translated from Trudy Nauchno-Tekhnicheskoi Konferentsii "Atomnaya Energetika, Toplivnye Tsikly, Radiatsionnoe Materialovedenie", Ul'yanovsk, 5--10 Oktyabrya 1970 G. Vol. II.

From conference on atomic power, fuel cycles, and radiation study of materials; Ul'yanovsk, USSR (5 Oct 1970).

In proceedings of the scientific-technical conference "atomic power, fuel cycles, radiation study of materials".

The calculated results of burnup of dysprosium with additions of gadolinium, erbium, and holmium oxides are given. The results of the effects of extended irradiation are discussed, and the stability and sizes of baked ceramic samples based on dysprosium oxides in a reactor at temperatures of 550 to 650°C and 750 to 850°C are studied. It is shown that the ceramic materials based on dysprosium oxides have a very high radiation stability. (auth)

NSA, vol. 32, 1975

38888 MECHANISM OF STRUCTURAL TRANSFORMATIONS IN BORON NITRIDE. Fel'dgun, L. I.; Nikitina, T. P.; Sokhor, M. I.; Futergendler, S. I. (All-Union Inst. of Abrasives and Grinding, Leningrad). Russ. J. Phys. Chem. (Engl. Transl.); 45: No. 12, 1738-40(Dec 1971).

Translated from Zh. Fiz. Khim.; 45: No. 12, 3067-70(Dec 1971).

A microstructure and x-ray-diffraction study was made on the products of the synthesis of cubic boron nitride from pyrolytic hexagonal boron nitride using a solvent melt. Under the given experimental conditions there is virtually no direct transition from hexagonal to cubic boron nitride. The importance of the role of the solvent melt in the phase transformation of boron nitride is noted. (auth)

NSA, vol. 26, 1972

25125 SINTERED NUCLEAR FUEL CONTAINING A BURNABLE NUCLEAR POISON AND PROCESS FOR ITS PREPARATION. (to Fiat SpA). Belgian Patent 743,232. 16 Dec 1969. Priority date 25 Feb 1969, Italy.

A sintered nuclear fuel, especially sintered uranium dioxide, containing a burnable nuclear poison is characterized by the fact that the burnable poison is uniformly distributed in the form of microspheres of 10 to 2000 microns in the nuclear fuel. (BE)

NSA, vol. 27, 1973

3,799,839

REACTIVITY AND POWER DISTRIBUTION CONTROL OF NUCLEAR REACTOR

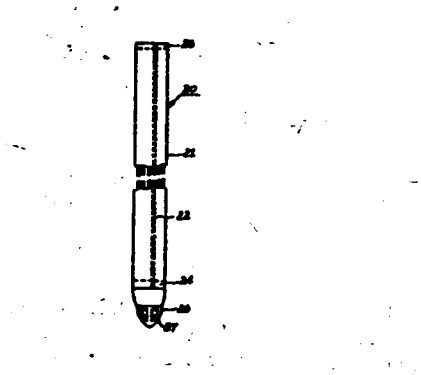
David L. Fischer and Frederick R. Channon, San Jose, Calif., assignors to General Electric Company

Continuation-in-part of application Ser. No. 762,776, Sept. 26, 1968, which is a continuation-in-part of application Ser. No. 708,391, Feb. 26, 1968, both now abandoned. This application Jan. 7, 1971, Ser. No. 104,705

Int. Cl. G21c *3/58*

U.S. Cl. 176—68

8 Claims



This describes a spatial distribution, amount, density and configuration of burnable poison to control a predetermined amount of excess reactivity and to maintain a constant or stationary power distribution during the operating cycle of a nuclear reactor core. In the illustrated embodiment of the invention the burnable poison is distributed throughout the core in a relatively small number of the fuel rods. In a preferred embodiment of the invention plutonium fuel is used in conjunction with the burnable poison to improve local power distribution and control margin and as an effective means for utilizing plutonium fuel.

U.S. PATENTS, 1974

27403 REACTIVITY AND POWER DISTRIBUTION CONTROL OF A NUCLEAR REACTOR CORE. Fischer, David Lloyd; Channon, Frederick Robert; Clancey, Walter Robert; (and others) (to General Electric Co.). British Patent 1,253,912. 17 Nov 1971. Priority date 26 Feb 1968, United States.

BURNABLE POISONS—core of, control of heterogeneous power reactor reactivity and power distribution by
REACTORS, HETEROGENEOUS—core for power, control of reactivity and power distribution by burnable poison
REACTORS, POWER—core for heterogeneous, control of reactivity and power distribution by burnable poison (D.C.C.)

NSA, vol. 26, 1972

8555 POISON FOR NUCLEAR FUELS. Gilissen, R.; Flipot, A. J. (to Centre d'Etude de l'Energie Nucleaire). Belgian Patent 781,976. 12 Apr 1972. 6p. (In French).

A process is described for fabrication of a burnable poison, 3 $Dy_2O_3 \cdot 5Al_2O_3$ and of fuel pellets based on uranium oxide. (tr-auth)

NSA, vol. 28, 1973

28183 NUCLEAR FUEL CONTAINING LANTHANIDES. Flipot, A. J.; Gilissen, R. (to Centre d'Etude de l'Energie Nucleaire). Belgian Patent 728,885. 25 Feb 1969.

A procedure is described for the fabrication of nuclear fuels containing a burnable poison consisting of a compound of an element of the lanthanide group characterized by the fact that a small amount of a compound of one of the following elements is added to the constituents of the fuels: titanium, aluminum, magnesium, zirconium, niobium, chromium, vanadium iron, nickel, or copper. (BE)

NSA, vol. 27, 1973

26264 SINTERING BEHAVIOR OF BORON. German, R. M. (Sandia Labs., Livermore, CA); Mar, R. W.; Hastings, J. C. Amer. Ceram. Soc. Bull.; 54: No. 2, 178-181 (Feb 1975).

From 26th Pacific coast regional meeting of the American Ceramic Society; San Francisco, CA (1 Nov 1973).

The sintering behavior of crystalline boron was investigated over the 2155 to 2268 K range and the effects of temperature, time and particle size were examined. Sintering progress was monitored by density and specific surface area changes. No densification took place upon sintering, but there were marked changes in specific surface areas and microstructural morphology. Results are discussed in terms of the surface diffusion and evaporation-condensation mechanisms. Sintering behavior is interpreted in light of the low-atomic bonding symmetry of boron. (auth)

NSA, vol. 31, 1975

48317 (HEDL-TME-72-88) FABRICATION MICROSTRUCTURE AND THERMAL CONDUCTIVITY IN UO_2–25 WT. PERCENT PuO_2 SINTERED PELLETS. Gibby, R. L. (Hanford Engineering Development Lab., Richland, Wash.). May 1972. Contract AT(45-1)-2170. 44p. Dep. NTIS.

The effect of pore structure on the thermal conductivity of 90% dense UO_2^- (20 to 25 wt %) PuO_2 fuel pellets was determined at temperatures from 700 to 1500°C. Thermal conductivity was determined from laser heat pulse thermal diffusivity measurements. Thermal conductivity data for samples with closed, isometric pores were found to agree with an established thermal conductivity design equation for mixed-oxides and with the modified Loeb equation recommended for porosity corrections in UO_2. Samples with laminar, open porosity had thermal conductivities as much as 33% less than the design equation while samples with microcrack networks had thermal conductivities as much as 44% less than the design equation. Thermal conductivities for all the samples obeyed the relationship $\lambda_e = (1 - P/1 + \beta P)\,[(1/2.88 + .0252T) + 5.82 \times 10^{-13}T^3]$ to ±10% where T is in °K, P is volume fraction porosity, and β is a pore shape factor. The pore shape factor varied from $\beta = 1.25$ for samples with closed, isometric pores to $\beta \simeq 10$ for samples containing dense microcrack networks. Application of the laboratory thermal conductivity data to refined thermal performance analyses for in-reactor experiments is discussed. Results of an irradiation experiment measuring relative melting heat ratings using fuels of different microstructures (conductivities) were found to be consistent with laboratory measurements. (auth)

NSA, vol. 26, 1972

37830 (BAW-3647-24) PHYSICS VERIFICATION PROGRAM, PART III. TASKS 5, 6, AND 7. Summary Report. Baldwin, M. N.; Fairburn, G. T. (Babcock and Wilcox Co., Lynchburg, Va. Research and Development Div.). Mar 1972. Contract AT(11-1)-3109. 99p. Dep. NTIS.

The response of all-solid, self-powered, incore neutron flux detectors is correlated to the power density profiles of a critical assembly. The experimental assembly, a water-moderated, slightly enriched UO_2 lattice perturbed by Ag–In–Cd poison pins and dissolved boric acid, is arranged to mockup the fuel elements of a commercial pressurized water reactor. A plutonium lattice for obtaining data for verification of the analytical methods and models developed for the design of water-moderated, plutonium-recycle cores is described. The lattice system is designed to provide a critical assembly with a small central test region and a nearly flat flux and asymptotic spectrum. The test region is incrementally poisoned with boric acid until a void substitution yields a null reactivity change. This provides a precise measure of the boron concentration required to reduce to unity the infinite-medium multiplication factor of the test lattice. Measurements of the effect of B_4C rods on the reactivity and power distribution in UO_2 lattices are described. (10 references) (auth)

NSA, vol. 26, 1972

1. Experimental Verification of Lumped Burnable Poison Models for PWR Cores, *I. D. Green, W. A. Wittkopf (B&W-Lynchburg)*

A series of experiments on critical assembly mockups of current and advanced PWR cores[1] was conducted at Babcock & Wilcox, and the results were compared with power distributions and lumped burnable poison rod worths obtained from PDQ07[2] calculations. The mockups (15 in all) contained various arrangements of water holes and lumped burnable poisons which simulate typical pressurized water reactor cores. Three types of lumped burnable poison rods were used to represent different degrees of burnup. Partial lumped burnable poison loadings were considered.

Figure 1 shows the core configuration for loading 8—one of the 15 experimental mockups. The core contains 4961 aluminum-clad fuel rods of uranium oxide enriched to 2.46 wt% ^{235}U. The midplane relative power distributions measurements were made in the central 15 × 15 pin subassembly. The excess reactivity was controlled by dissolving boric acid in the moderator.

Current design methods were used for the analytical calculations of the critical assembly mockups. PDQ07 was used for the numerical solution of the diffusion equations with the cross sections from the LIFE[3] program. Three lumped burnable poison models were considered: (a) a detailed model which has four mesh lines per pin cell and infinite-medium cell cross sections; (b) a discrete model which has one mesh line per pin cell, infinite-medium cell cross sections, and a modified Σ_a absorption of the lumped burnable poison cell to correct for diffusion theory tendency to overestimate absorption in a high absorber region; and (c) a homogenized two-zone model.

Table I compares the calculated eigenvalues and the lumped burnable poison rod cluster worths for the three lumped burnable poison models. The rod cluster worths are calculated in ppm of soluble boron to compare directly with experiment. Relative power densities were measured in each fuel pin and showed good agreement with the discrete and the detailed calculational models. Power densities in fuel pins adjacent to lumped burnable poison pins are within ±1.5% of calculated densities.

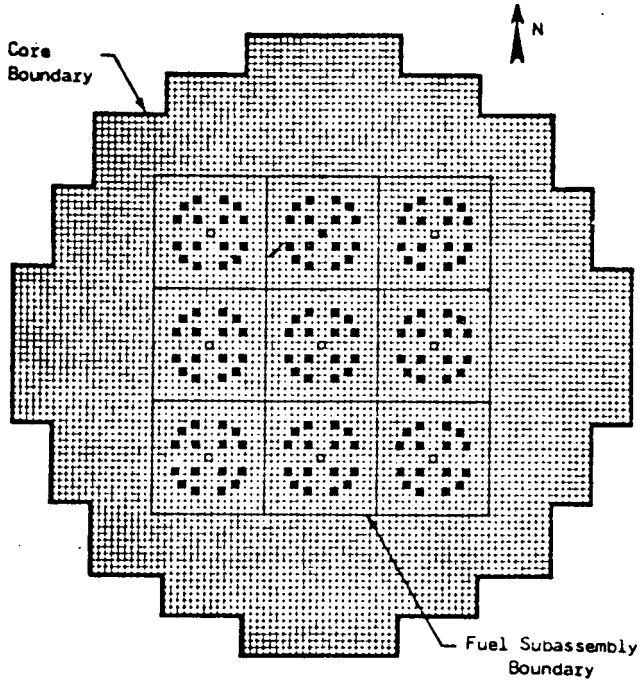


Fig. 1. Loading 8 core configuration.

1. M. N. BALDWIN and M. E. STERN, "Physics Verification Program, Part III, Task 4, Summary Report," BAW-3647-20, Babcock & Wilcox (March 1971).
2. W. R. CADWELL, "PDQ07 Reference Manual," WAPD-TM-678, Bettis Atomic Power Laboratory (January 1967).
3. W. A. WITTKOPF et al., "LIFE—A Nuclear Design Package Program," BAW-TM-402, Babcock & Wilcox (February 1968).

(Cont'd pg 151)

TABLE I

Calculated Eigenvalues and Rod Cluster Worths

Loading	Calculated Eigenvalues			Rod Cluster Worth in ppm Soluble Boron			
	Detailed	Discrete	Two-Zone	Experimental	Detailed	Discrete	Two-Zone
1	0.99575	0.99565	0.99594				
2	0.99396	0.99569	0.99726				
3	0.99397	0.99573	0.99680				
4	0.99285	0.99445	0.99471	151	159	159	166
5	0.99252	0.99413	0.99546	153	163	163	163
6	0.99289	0.99376	0.99649	300	307	311	303
7	0.99224	0.99358	0.99896	303	315	316	289
8	0.99451	0.99391	1.00101	540	536	549	513
9	0.99416	0.99355	1.00304	558	557	570	517
10	0.99431	0.99666	0.99631	89	87	82	93
11	0.99431	0.99610	0.99551	-50	-52	-52	-41
12	0.99376	0.99552	0.99385	-14	-13	-14	7
13	0.99376	0.99552	0.99394	-14	-13	-14	6
14	0.99347	0.99525	0.99390	-29	-26	-27	-8
15	0.99352	0.99530	0.99412	-28	-25	-26	-9
$\bar{\lambda}$	0.99373	0.99499	0.99649				
$\pm\sigma$	±0.00088	±0.00096	±0.00269				

ANS TRANS., vol. 15, 1972

20747 KINETICS OF HETEROGENEOUS THERMAL DECOMPOSITION OF ZIRCONIUM AND HAFNIUM BORON HYDRIDES. Grigor'ev, V. A. (Inst. of Inorganic Chemistry, Novosibirsk, USSR). Kinet. Katal.; 12: No. 4, 1042-5(Jul-Aug 1971). (In Russian).

Samples of $Zr(BH)_4$ and $Hf(BH)_4$ were synthesized and used in a study of the kinetics of their decomposition over the temperature range 200 to 300°C. The variables controlled were: temperature, time, total gas pressure, and amount of nonvolatile material remaining on the surface of the sample. The rate constants of the heterogeneous decomposition, the energy of activation, and the enthalpy of activation were measured. (K.S.W.)

NSA, vol. 26, 1972

21912 APPLICATION OF CADMIUM COMPOUNDS AS A BURNABLE POISON IN NUCLEAR REACTORS. Hattenbach, K.; Melkonian, G. A.; Baur, E. (to Gesellschaft fuer Kernenergieverwertung in Schiffbau und Schiffahrt mbH). German Patent 1,764,164. 17 Feb 1972. (In German).

The claim deals with the application of a spinel-like mixed oxide, composed of cadmium and an additional metal according to the formula $Me(I)Me_2(II)O_4$ using cadmium aluminate ($CdAl_2O_4$) or cadmium zirconate (Cd_2ZrO_4) as a burnable poison. (GF)

NSA, vol. 27, 1973

3674 SEQUOYAH REACTORS: FUEL AND FUEL COMPONENTS. Heil, August L. (Westinghouse Electric Corp., Pittsburgh). Nucl. Eng. Int.; 16: No. 185, 857-9(Oct 1971).

Details are presented of the fuel assemblies and the fuel rods, which consist of UO_2 in Zircaloy-4. Fuel assembly fabrication is outlined. Features adopted to increase reliability and minimize fuel cycle costs are described. They include the use of chemical shim control, rod cluster control, burnable poison in the initial core cycle, and prepressurized fuel rods. Non-uniform loading of fuel is maintained by a cycled core-management technique. (UK)

NSA, vol. 26, 1972

30450 HETEROGENEOUS METHOD FOR CALCULATING LONG TERM REACTIVITY CHANGE IN A REACTOR INCLUDING BURNABLE POISON RODS. Hishida, Hisashi (Mitsubishi Atomic Power Industries, Inc., Omiya, Japan); Sekiya, Tamotsu. Nucl. Sci. Eng.; 47: No. 3, 319-28(Mar 1972).

A heterogeneous method of calculating time-dependent reactor core characteristics, such as the time variation in thermal-neutron flux distribution and the reactivity change during fuel and poison burnup, is derived. The lattice consists of an infinite number of similar square zones closely connected to one another. In each zone, identical fuel rods are arranged in a regular lattice with a burnable poison rod of the same geometric dimensions as a fuel rod at the center. Some numerical examples, utilizing the equations derived finally, give the time variation in poison concentration and $k_\infty(t)$ for a zone showing the heterogeneity effect associated with a burnable poison rod. Since the machine time required to compute the time variation of such core characteristics through fuel life of 11,000 EFPH as shown in the examples is <25 sec on the IBM 360/75 per case, the method may be applied to the preliminary survey calculation for the time-dependent heterogeneous core characteristics of a square lattice including burnable poison rods as well as to more general time-dependent problems related to such lattices. 9 references. (auth)

NSA, vol. 26, 1972

45120 INVESTIGATIONS OF EUROPIUM OXIDE CERMETS FOR CONTROL ELEMENTS IN NUCLEAR REACTORS. Hoffmann, A.; Pant, P. (Fried-Krupp GmbH, Essen). Tech. Mitt. Krupp, Forschungsber.; 29: No. 1, 15-25(May 1971). (In German).

Europium molybdate and europium titanate as well as sintered parts from metallic powders and powders containing europium oxide were produced. These sintered parts were corrosion tested in water at 250°C. Cermet plates containing 40.75 to 51% Eu_2O_3 in the core were produced. Metallographic and radiographic examinations of the cermet plates were carried out. (auth)

NSA, vol. 25, 1971

15621 INVESTIGATIONS OF EUROPIUM OXIDE CERMETS WITH A VIEW TO THEIR USE FOR CONTROL RODS IN NUCLEAR REACTORS. Hoffmann, A.; Pant, P. (Krupp Friedr. GmbH, Essen). Tech. Mitt. Krupp, Forschungsber.; 29: No. 2, 15-25 (1971). (In German).

Results of investigations of the preparation of europium mixed oxides with MoO_3 and TiO_2 and of sintered specimens of europium mixed oxides and V2A- or V4A-powders, respectively, are reported along with those from investigations of the microstructures and of the corrosion resistance of these cermets in water. The manufacture of plated cermet kernels with europium oxide contents between 40.75 and 51.0% is described. The perfect structure of the prepared cermet plates could be demonstrated by means of metallographic and x-ray investigations. (INIS)

NSA, vol. 26, 1972

32031 (A/CONF.28/P/253) CONTROL ROD MATERIALS. Abe N. Holden (General Electric Co. Vallecitos Atomic Lab., Pleasanton, Calif.), Bernhardt Weidenbaum, and Carl F. Leitten, Jr. 16p.

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

A review is presented on the fabrication, physical properties, radiation effects and performance of boron materials, Hf, Ag–In–Cd, and lanthanons, for use as control rod materials. Materials used for control rods in U. S. power reactors are listed, and data on boron compounds are tabulated. The economics of control rod materials is briefly discussed. (P.C.H.)

NSA, vol. 18, 1964

6161 REACTOR CORE AND FUEL ELEMENTS OF THE NUCLEAR POWER PLANT STADE. Holzer, R.; Knoedler, D. (Siemens AG, Erlangen, Ger.). Atomwirt., Atomtech.; 16: No. 11, 581-4(Nov 1971). (In German).

In comparison with the nuclear power plant Obrigheim, core and fuel elements of the nuclear power plant Stade show advanced development with respect to enhancement of absolute as well as specific power and burnup, increased operational safety, improved accident protection, easier fuel element exchange, easier access for inspection, and simplification of manufacturing and quality control. Briefly discussed are changes in core assembly, poisons, neutron sources, and fuel elements. (tr-auth)

NSA, vol. 26, 1972

11461 (IN-1437, pp 13-23) PROTOTYPE ELEMENTS. (Idaho Nuclear Corp., Idaho Falls).

From Metallurgy and Materials Science Branch annual report, fiscal year 1970.

Results of tests on B_4C-impregnated Al foils as controllable neutron absorbers for fuel elements, are presented. Investigations of core breakup during fabrication of ETR fuel plates are summarized along with work devoted to development of 2219 Al alloy as fuel cladding. (J.R.D.)

NSA, vol. 25, 1971

6249 (RT/FI-(71)9) CROSS SECTION LIBRARIES SETUP FOR BURNUP CALCULATIONS IN THE PRESENCE OF Gd AS A BURNABLE POISON IN LWR'S. Iorio, G.; Pistella, F.; Sisto, F. (Comitato Nazionale per l'Energia Nucleare, Rome (Italy)). 1971. 41p. Dep. NTIS (U. S. Sales Only).

A cross section library has been set-up to be used for burnup calculations in the presence of Gd. The available cross section data are briefly discussed, a new reduced energy scheme is suggested and its validity is tested, the values resulted for the materials of interest are also given. A few assumptions are discussed which allow a simplified model to be used in burnup calculations. 12 references. (auth)

NSA, vol. 26, 1972

18577 WAPD-221

Westinghouse Electric Corp. Bettis Atomic Power Lab., Pittsburgh.

DEVELOPMENT OF BORON-BEARING MATERIALS FOR APPLICATION AS LUMPED BURNABLE POISONS.

D. Jaffe, E. F. Losco, and S. F. Kaufman. May 1960. 66p. Contract AT-11-1-GEN-14. OTS.

Several iron, stainless steel, and niobium-base boron-bearing materials containing 1 to 3 wt.% B^{10} were investigated for possible applications as lumped burnable poisons. Fabrication methods, mechanical and physical properties, and corrosion resistance were examined. On the basis of out-of-pile evaluation and some preliminary capsule irradiation data, the most promising materials were austenitic (Fe + 18 wt.% Cr + 15 wt.% Ni) and ferritic stainless steel (Fe + 18 wt.% Cr) alloys with less than 2 wt.% B^{10}. Cladding of these materials plus some additional iron and stainless steel base alloys with Zircaloy-2 was accomplished by roll bonding, using intermediate barrier materials. Sub-sized Zircaloy-2 clad poison plates were prepared for in-pile loop tests by roll bonding and compartmentation. Final selection of a poison material will be dependent on the results of the in-pile tests. (auth)

NSA, vol. 14, 1960

3,781,191

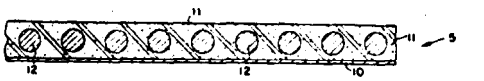
NUCLEAR CONTROL ROD HAVING EMBEDDED FILAMENTS

Cecil R. Jones, Hamden, Conn., assignor to Transfer Systems Incorporated, New Haven, Conn.

Filed Feb. 22, 1972, Ser. No. 228,067

Int. Cl. G21c 7/10

U.S. Cl. 176—86 R 8 Claims



A nuclear reactor control rod for regulating or shutting-down the power of a nuclear reactor is described. The control rod features filaments of a neutron absorber bonded together and to a foil backing to form a sheet. Plural sheets are laminated to form a plate, and plural plates may be joined together in a cruciform shape. A preferred method to make such a control comprises forming the sheets by plasma-spraying a corrosion-resistant metal to bond the filaments to the foil backing.

U.S. PATENTS, 1973

6. Fuel Management with Lumped Burnable Poison, *Henry M. Jones, Steve Thompson, J. S. Tulenko (B&W-Lynchburg)*

Currently lumped burnable poison (LBP) in the form of boron is used in most first cycles of PWRs as a means to limit the BOL moderator coefficient. This paper reports on the extension of the current application of lumped burnable poison to its use as an additional design parameter for in-core fuel management.

Studies[1-3] have been completed and demonstrate that the use of LBP clusters in judiciously chosen fresh fuel loading patterns can result in significant uranium enrichment savings for PWR uranium core designs while meeting core power distribution requirements. The use of lumped burnable poison was shown to result in a 3 to 5% reduction in feed enrichments while maintaining batch discharge exposures. The net fuel cycle cost savings obtained after adjustment for the LBP material and fabrication costs is approximately 0.2¢ per million BTU. Additionally, studies[2,3] have shown that significant fuel management flexibility to accommodate fuel cycle modifications is achieved by the utilization of LBP.

The fuel management scheme which utilizes the LBP clusters in selected fresh fuel assemblies places the fresh fuel in the inner two-thirds of the core on a checkerboard array with twice-burned fuel and places the once-burned fuel in the outside one-thrid region of the core volume. The LBP pins contain a uniform concentration per cluster of 16 pins. These clusters are placed in the empty control rod guide tubes of the fresh fuel to control the power distribution and remain in that fuel for one cycle. The essential feature of the use of LBP in fuel management is the conservation of neutrons over cycle lifetime by reduced core radial leakage counterbalanced by an increased power level in the twice-burned fuel. This aspect is graphically shown in Figs. 1 and 2 which show a comparison of the average power distribution in each of six radial zones at beginning- and end-of-life for both an LBP and an out-in fuel management scheme. The average power in the high leakage outer zone of the out-in scheme can be seen to be 40% greater than that for the LBP scheme in the same zone. The average power in the hot pin is essentially identical at BOL for the two cases; i.e., 1.388 for the out-in scheme versus 1.391 for the LBP scheme. An examination of the power distribution at end of cycle shows that the power peak remains virtually unchanged for the LBP scheme over lifetime (1.40) while it decreases to 1.30 for the out-in scheme.

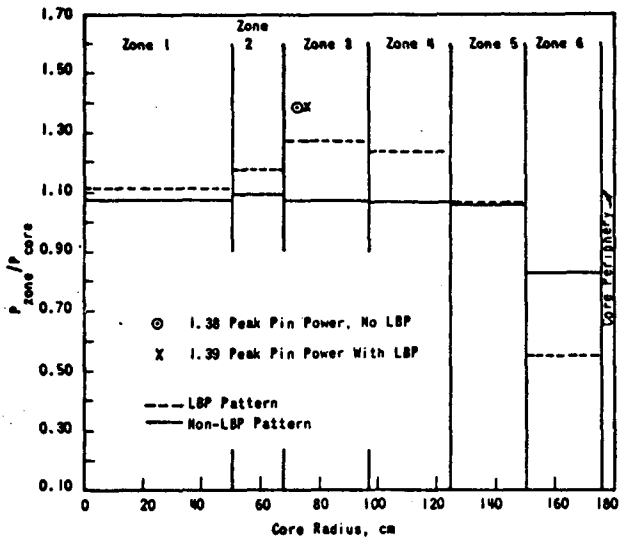


Fig. 1. Average power in each of six radial zones normalized to core average power-typical Cycle-40 EFPD operation, BOL.

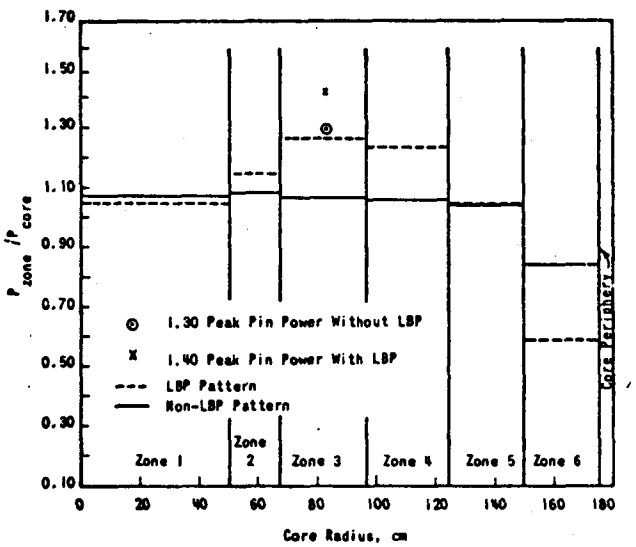


Fig. 2. Average power in each of six radial zones normalized to core average power-typical Cycle-310 EFPD operation, EOL.

1. F. M. ROSS et al., "Fuel Shuffle Analysis For The 205 Fuel Assembly Core," NPGD-TM-81, Babcock & Wilcox (March 1970).
2. P. H. KLINK and J. K. ARTHUR, "Study Of Flexibility—205 Fuel Assembly Core," NPGD-TM-109, Babcock & Wilcox (October 1970).

4067 (FEI-220) BURNUP OF CYLINDERS OF PURE ABSORBER. Kalashnikov, A. G.; Toshinskii, G. I.; Tsurkova, E. V. (Gosudarstvennyi Komitet po Ispol'zovaniyu Atomnoi Energii SSSR, Obninsk. Fiziko-Energeticheskii Institut). 1970. 17p. (In Russian). INIS.

If it is possible to select a burnup absorber with a sufficiently high rate of burnup, then spacing of this absorber in the form of cylinders with large diameters appears to be the most efficient. A method for accurate calculation of burnup of cylindrical rods of depleted absorber containing pure absorber under the influence of a known incident neutron flux is presented. A computer program for calculating cadmium and gadolinium rods is briefly described. The results from such a program may be used for testing various burnup models in reactor calculations. (tr-auth) (INIS)

NSA, vol. 26, 1972

11214 CONTROL OF BWR POWER DISTRIBUTION. Kilian, P.; Pohl, P. (Kraftwerk Union A. G., Offenbach (F.R. Germany)). Atomkernenergie; 25: No. 2, 103-104(1975). (In German).

4 figs.; 6 refs.

In a boiling water reactor, the power distribution can be shaped by proper use of the control rods. Besides that they have to balance the effect of variable void content and, in the long run, to compensate burnup reactivity. In operation a constant target power distribution is maintained by multiple control rod configurations. Operation of the reactor core is assisted by the nuclear model process computer which constructs the power distribution from the output of the incore detectors. (auth)

NSA, vol. 33, 1976

14545 EXPERIMENTS FOR THE USE OF PLATES AS SHIM RODS IN REACTORS WITH MTR-TYPE FUEL ELEMENTS. Krull, W. (Gesellschaft fuer Kernenergieverwertung in Schiffbau und Schiffahrt mbH, Geesthacht, Ger.). Atomkernenergie; 18: No. 3, 198-202(Nov 1971). (In German).

Flux density and reactivity measurements of MTR elements with plate-type shim rods have been performed. The influence on the horizontal thermal neutron flux density distribution and on the reactivity worth of the position of the plate-type shim rods in the special fuel elements was measured. The result is that the distance between the two plates should be as large as possible. The reactivity worth of plate-type shim rods made of Ag-In-Cd (80-15-5) is 85% higher than for the old oval-type shim rods. For Cd-plates the dependence on the reactivity worth on Cd thickness was measured. Ag-In-Cd plates have a 16.5% higher reactivity worth than pure Cd-plates. The results give a clear indication for the optimum design of plate type shim rods for MTR-fuel elements. (7 references) (auth)

NSA, vol. 26, 1972

28169 (RT/FI-73-41) EXPERIMENTS ON THE USE OF Gd_2O_3 AS BURNABLE POISON FOR PWR. Final Project of Sample Irradiation in Siloe. Lepscky, C.; Pistella, F. (Comitato Nazionale per l'Energia Nucleare, Rome (Italy)). 1973. 52p. (In Italian). INIS.

7 figs.; 8 tables.

The results are reported of calculations performed to define the set-up of the experiments conducted by nuclear ship program of CNEN in the SILOE reactor (CEA Grenoble), which consist of irradiating natural UO_2 samples containing Gd_2O_3 in different concentrations. (INIS)

NSA, vol. 33, 1976

5. Effect of Boron and Gadolinium on the Criticality of Plutonium-Uranium Systems, *R. C. Lloyd, E. D. Clayton (Battelle-Northwest)*

Experiments were performed to substantiate the effectiveness of soluble nuclear poisons on the criticality of plutonium-uranium nitrate solutions and to determine their effectiveness for criticality control on a heterogeneous assembly of mixed-oxide fuel pins immersed in plutonium-uranium nitrate solutions containing the soluble poisons.

Calculations indicate the most effecive use of soluble poisons may involve mixtures of several different elements, such as a mixture of boron and gadolinium, since one nuclide is more effective than another in absorbing neutrons from different portions of the neutron spectrum as the fuel concentration, or H/X ratio, is varied. As these are the first data known to be reported on boron-gadolinium mixtures, the data provide the only validation points for calculations of the mixed absorber effects.

Heterogeneous System: The lattice contained 301 fuel pins in a 55.5-cm-i.d. cylindrical vessel with 0.079-cm wall thickness. Pu+U nitrate solution was pumped into the vessel, using the critical approach method to determine the critical height. The Pu+U solution contained about 80 g Pu/liter. The ^{240}Pu content of the Pu was 6.3%, and the ^{235}U content of the U was 0.66%. The Pu made up about 30% of the total Pu+U by weight. The solution contained a combination of B and Gd in the ratio ~3-to-1 by weight. The data from these experiments are summarized in Table I.

Homogeneous System: The experiments were performed on a water-reflected, stainless-steel cylindrical vessel of 61.03-cm i.d. and wall thickness 0.079 cm. Critical experiment data on the homogeneous Pu+U solutions containing various concentrations of B and Gd are given in Table II. The variation in critical height with added quantity of neutron poison can be seen from the value given in the table. Several of the Pu+U solutions are the same solutions as used in the heterogeneous system.

The results of theory-experiment comparisons utilizing ENDF/B cross sections and the KENO Monte Carlo code will be presented and discussed.

(Cont'd pg 155)

TABLE II

Effect of Boron and Gadolinium on Criticality of Homogeneous Systems
(U + Pu Solutions in 61-cm-diam Water-Reflected Cylinder)

Date	Exp No.	Critical Height (cm)	Pu Conc (g/liter)	U Conc (g/liter)	Acid Molarity	Total NO_3 (g/liter)	Specific Gravity	Gd (g/liter)	B (g/liter)
4-7-75	137R	18.13	85.0	182.5	1.55	320	1.433	0.04	0.0
4-9-75	138	18.65	84.9	182.2	1.61	311	1.434	0.04	0.1
4-11-75	139R	19.68	84.8	182.6	1.67	290	1.433	0.04	0.3
4-17-75	140	21.13	84.5	182.1	1.75	296	1.438	0.128	0.27
5-20-75	145	32.16	82.8	180.6	1.95	309	1.443	0.293	0.9
5-22-75	146	43.03	82.2	179.5	2.12	319	1.446	0.388	1.2
5-26-75	147	52.12	81.2	180.0	2.18	321	1.447	0.424	1.35
5-16-75	150	67.83	81.0	180.4	2.09	316	1.451	0.519	1.5
7-11-75	151	75.44	81.0	180.3	2.21	318	1.452	0.537	1.54

TABLE I

Effect of Boron and Gadolinium on Criticality of Heterogeneous Lattice Assembly

Date	Exp No.	Critical Height (cm)	Pu Conc (g/liter)	U Conc (g/liter)	Acid Molarity	Total NO_3 (g/liter)	Specific Gravity	Gd (g/liter)	B (g/liter)
5-7-76	141R	21.20	84.5	182.1	1.75	296	1.438	0.13	0.27
5-9-75	142	25.37	84.0	183.2	1.88	302	1.442	0.235	0.6
5-13-75	143	30.49	82.7	180.6	2.01	308	1.444	0.309	0.9
5-30-75	148	43.86	81.2	180.0	2.18	321	1.447	0.424	1.35
6-3-75	149	51.97	81.0	180.4	2.09	316	1.451	0.519	1.5
7-17-75	152	55.18	81.0	180.3	2.21	318	1.452	0.537	1.548
7-18-75	153	65.42	80.5	180.5	2.24	321	1.454	0.541	1.662

Fuel Pin Dimensions			Fuel Per Pin
	o.d. (cm)	Length (cm)	PuO_2-U (nat)O_2: 138.4 ± 1.3 g
Fuel column	0.495	69.22	Pu: 30.75 ± 0.03 g
Cladding (316-SS)	0.584	72.90	U: 91.16 ± 1.03 g
			O: 16.49 ± 0.17 g

Fuel Enrichment	Fuel Density
25.2 wt% Pu	10.35 ± 0.09 g/cm^3 (93.34 ± 0.79% theoretical)

Isotopic Composition of Pu in Pins	Lattice
^{238}Pu: 0.04 ± 0.01 at.%	Triangular spacing = 3.048 cm
^{239}Pu: 86.19 ± 0.06 at.%	
^{240}Pu: 11.88 ± 0.06 at.%	
^{241}Pu: 1.73 ± 0.01 at.%	
^{242}Pu: 0.16 ± 0.01 at.%	

17355 (BNWL-tr-44) BORON CARBIDE: PRODUCTION, PROPERTIES, AND APPLICATIONS. Lipp, Alfred. Translated for Battelle-Northwest, Richland, Wash., from Tech. Rundsch.; Nos. 14, 28, 33(1965); No. 7, 3-47(1966). 101p. Dep. CFSTI.

A review of information on production, properties, and uses of boron carbide is presented along with information on crystal structure. Phase diagrams are also included. (J.R.D.)

NSA, vol. 24, 1970

36522 PHYSICOCHEMICAL INTERACTION OF SAMARIUM AND GADOLINIUM OXIDES WITH BARIUM OXIDE. Lopato, L. M.; Maister, I. M.; Shevchenko, O. V. (Inst. of Problems in Material Management, Kiev). Dopov. Akad. Nauk Ukr. RSR, Ser. B; No. 3, 245-8(Mar 1972). (In Ukrainian).

Phase diagrams of the systems Sm_2O_3-BaO and Gd_2O_3-BaO in the concentration range from 0 up to 90 mol. % BaO and within the temperature interval of 1200 to 2400°C were plotted by the method of annealing and tempering as well as by means of the high-temperature differential thermal analysis. It was established that in the studied systems there exist two types of compounds: that of composition 1:1 (structure $CaFe_2O_4$) and that of composition 2:3 (hexagonal structure). In the systems Eu_2O_3-BaO and Tb_2O_3-BaO the compounds of 1:1 and 2:3 composition were prepared as well. The effect of barium oxide additives was investigated as applied on the character of temperature change in polymorphous transformations A ⇌ H, H ⇌ X, B ⇌ A of samarium and gadolinium oxides. (auth)

NSA, vol. 26, 1972

14469 (BAW-3647-23) PHYSICS VERIFICATION PROGRAM. PART III, TASKS 5 AND 7. Quarterly Technical Report, July-September 1971. Baldwin, M. N. (Babcock and Wilcox Co., Lynchburg, Va. Research and Development Div.). Dec 1971. Contract AT(30-1)-3647. 41p. Dep. NTIS.

An experimental task to correlate the response of all-solid, self-powered, incore neutron flux detectors to power density profiles of a critical assembly is described. The experimental assembly, a water-moderated, slightly enriched, UO_2 lattice perturbed by Ag-In-Cd poison pins and dissolved boric acid, is arranged to mock up fuel elements of a commercial pressurized water reactor. Results obtained during the third calendar quarter are presented. They will serve to check analytical techniques for relating detector signals and power profiles in commercial power reactors. Also described is a second experimental task utilizing the same basic core lattice to obtain measurements of the effect of B_4C rods on the reactivity and power distribution of the assembly. Except for the B_4C poison rods, these lattice loadings are identical to cores similarly mapped and reported earlier. Since the earlier loadings were perturbed by Ag-In-Cd poison rods, a direct comparison of the two commonly used control rod materials, B_4C and Ag-In-Cd, is available. (7 references) (auth)

NSA, vol. 26, 1972

13053 PRODUCTION OF BURNABLE POISON OXIDE FUEL. Littlechild, J. E.; Butler, G. G.; Lester, G. W. (BNFL, Springfields, Eng.). pp 65.1-65.4 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004-.

In recent years there has been increasing consideration and application of burnable poison fuels in nuclear reactors, particularly in LWR's. This type of fuel enables a more constant core reactivity with time to be achieved by using high neutron capture cross section additives which absorb neutrons and revert to lower cross section nuclides at a similar rate to that at which the overall fissile reactivity is dropping due to burnup. The main criteria for the selection of burnable poisons are a neutron capture cross section which will ensure burnout at the correct rate, the absence of objectionable daughter products after irradiation, and the compatibility of the burnable poison and any daughter products with the fuel and clad under reactor operating conditions. (auth)

NSA, vol. 32, 1975

9845 PREPARATION OF SAMARIUM, GADOLINIUM, AND DYSPROSIUM NITRIDES. Lyutaya, M. D.; Goncharuk, A. B. J. Appl. Chem. USSR (Engl. Transl.): 44: No. 6, 1426-9(Jun 1971).

Translated from Zh. Prikl. Khim.; 44: No. 6, 1410-12(Jun 1971).

Two methods were studied for the preparation of Sm, Gd, and Dy nitrides. The results showed that in order to obtain nitrides of stoichiometric composition, nitriding of Sm, Gd, and Dy by the action of N must be conducted at temperatures close to the melting points of the metals. When ammonia was used as the nitriding medium, preparation of Sm, Gd, and Dy nitrides in a short reaction time (1 hr) also required the use of high temperatures, at which the hydrides formed were converted completely into nitrides. (P.C.H.)

NSA, vol. 26, 1972

Calculation of LWR Fuel Elements Containing Burnable Poisons and Plutonium, *C. Maeder, J. M. Paratte (EIR–Switzerland)*

INTRODUCTION

It is not yet possible to calculate a light-water reactor by considering all its geometrical details at the same time. The core is therefore divided into a number of elementary regions. Homogenized group constants for these regions are derived and used for calculation of the complete reactor. From the power distributions within the elementary regions and the whole reactor calculation, the hottest fuel element may be determined. This power should not exceed a critical value above which the reactor could be damaged.

The elementary regions often have a rather heterogeneous structure. Figure 1 shows a BWR fuel element as an example. Recently, fuel rods containing the burnable poison, gadolinium, have been introduced into the boiling water reactors, and plutonium recycling is in sight. These isotopes have large cross sections and therefore increase the heterogeneous nature of the fuel elements.

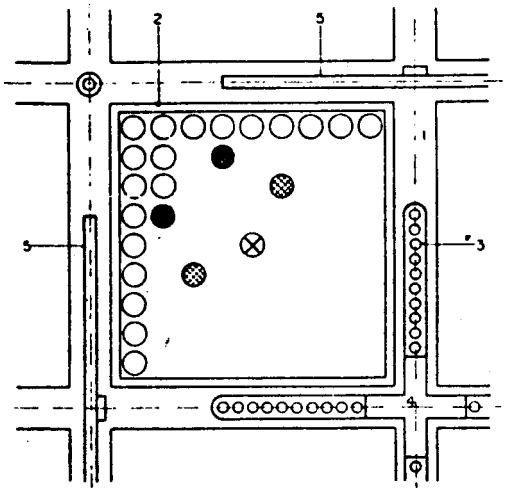


Fig. 1. BWR fuel element.

Calculation errors in the flux distribution within a fuel element are amplified during burnup.[1] The corresponding uncertainty in the fuel lifetime must be compensated by an increase in enrichment. Thus, economy and safety reasons require reliable calculation of the fuel elements, although their heterogeneity makes this task difficult.

DESCRIPTION OF THE BOXER PROGRAM

Input and Cross Sections

Figure 2 is a flow diagram of the EIR fuel element program BOXER (BOX-EIR).

The input is read in the program DEFINE, which requires only the modifications of the data from the previous case.

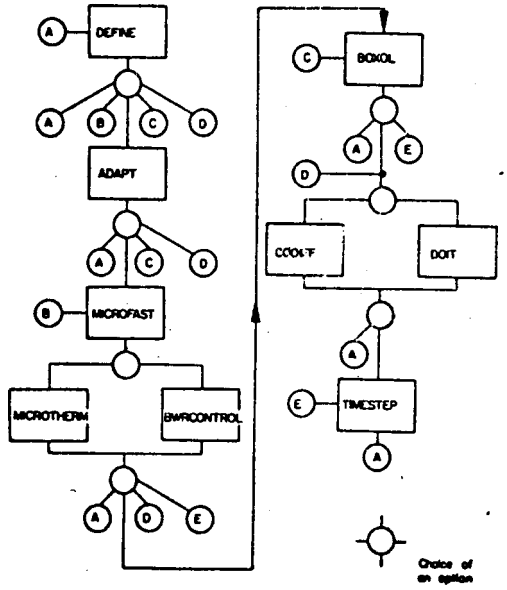


Fig. 2. Flow diagram of the BOXER program.

ADAPT prepares the microscopic temperature-dependent cross sections for cell calculations. It needs a cross-section file as input that has to be generated in advance from ENDF/B data by means of a program not belonging to BOXER. In particular, ADAPT determines the Doppler broadening of the resonances and thermal scattering matrices with the free gas and the Nelkin models.

Cell Calculations

The fuel elements are divided into homogeneous and heterogeneous materials. Examples of homogeneous materials are the water between the boxes and their Zircaloy sheaths. The heterogeneous materials have a fine structure, such as a cylindrical fuel cell (fuel rod, cladding, and moderator) or the control rod. Some typical configurations are selected among the heterogeneous materials, and fine structure calculations are performed for these typical cells only. The most representative cell of the element is calculated first with a zero-current boundary condition. The resulting outward boundary current is then used as an external source for the remaining cells. This source represents the surrounding material in an approximate manner.

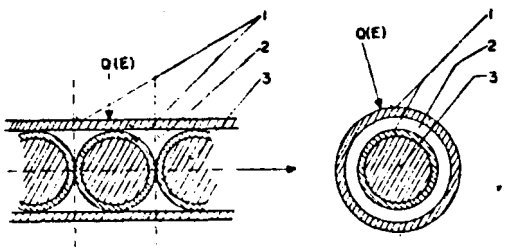
(Cont'd pg 158)

The fast energy range (no upscattering) is calculated by MICROFAST. This program is based on the interface current technique[2] with isotropic scattering. In space, the neutron flux is expanded into a Legendre polynomial series. The energy scale of the resonance region is divided into a great number of points (a typical value is 2000 energy points), which are selected by the program according to the behavior of the cell-averaged total cross section.

The thermal fuel cell spectrum in NGT1 = 15 to 30 energy groups, is determined with MICROTHERM. This program can consider anisotropic scattering which may be important in gadolinium and plutonium cells. The calculation method adopted in MICROTHERM is a mixture of integral transport and P_N theory, together with a space polynomial expansion of the neutron flux.[3] No inner iterations are needed within each group, since the flux equations are solved by the faster and more accurate direct method.

The microcell of a BWR control rod is first transformed into a cylindrical cell for which a MICROFAST calculation is performed (Fig. 3). In this manner, condensed group constants for the resonance region are obtained. The correct 2-D geometry is then considered in a calculation with BWRCONTROL, whereby the whole energy range is divided into NG1 = 50 to 100 energy groups. BWRCONTROL again uses an interface current method[1] with a DP_0 or a DP_1 approximation of the angular flux at the zone boundaries. Furthermore, isotropic scattering and a constant neutron source in each zone are assumed.

For the homogeneous materials, a zero-dimensional calculation is performed with a neutron source whose spectrum is given by the external source used in the cell calculations.



------planes of symmetry

1 boron carbide

2 stainless steel

3 water

Fig. 3. Control rod cells.

Approximate Two-Dimensional Geometry

With the help of the calculated flux distributions, cross sections are generated for each material in NG2 = 20 to 50 energy groups. NG2 should be rather large because of the approximate nature of the above-mentioned external source. Calculation of the whole fuel element is now performed with the BOXOL program. It considers each material as one zone only and uses approximate collision probabilities. Because of these approximations, the BOXOL results are not accurate. Nevertheless, they contain the influence of all materials occurring in the element on the flux spectrum. By means of the BOXOL fluxes, the cross sections are condensed into NG3 = 5 to 10 energy groups for the succeeding fine structure calculation.

Flux Distribution in the Fuel Element

At this point the program user can choose between the diffusion code CODIFF and the transport code DOIT. CODIFF is a further development of the program described in Ref. 4. The DOIT method divides the sphere formed by the neutron flight directions into subdomains, similar to the widely used S_N programs, but the use of discrete angular directions is avoided. The result is a mitigation of the ray effect and a better rate of convergence.[5,6]

Evolution of Isotopic Densities

The time variation of the isotopic densities in the materials of the fuel element is calculated by the TIMESTEP program. Every heterogeneous material is decomposed into its constituents by means of the flux distribution obtained in the corresponding cell calculation. Thus, the isotopic density may be radially dependent within a fuel rod. The differential equations for the isotopic densities are integrated by expanding them into Taylor series in the time variable.

CONCLUSIONS

The user of BOXER can proceed in a number of different ways through the program (Fig. 2). Every step can be omitted if the results of a previous run are still sufficiently accurate. Furthermore, several parameters can be varied within each program itself, as, e.g., the degree of space polynomial approximation in MICROFAST and MICROTHERM, the number of basic angular functions in MICROFAST and BWRCONTROL, the degree of anisotropy in MICROTHERM and DOIT, and the number of angular domains in DOIT. Thus, the user of BOXER can determine the accuracy of his calculation by himself provided he is willing to pay the corresponding price.

1. J. M. PARATTE, "La Base en Gestion du Combustible: un Code de Boîtes Moderne," to appear in *Neue Technik*, 16, *11* (1973).

2. M. M. ANDERSON and H. C. HONECK, "An Interface Current Technique for Two-Dimensional Cell Calculations," CONF-7304 14, Vol. 1, p. I-53, USAEC (1973).

3. J. LIGOU, "Improved Integral Transport Theory by Means of Space Polynomial Approximations," *Nucl. Sci. Eng.*, 50, 135 (1973).

4. J. M. PARATTE, "DIFFUS, un Programme basé sur la Théorie de Diffusion pour le Calcul à Deux Dimensions des Flux Neutroniques," EIR-Report No. 237 (1973).

5. J. ARKUSZEWSKI, "New Spatial Differencing for the Neutron Transport Equation," EIR-Report No. 232 (1973).

6. C. MAEDER, "Modified Versions of the Two-Dimensional Transport Theory Method DOIT," EIR-Report No. 240 (1973).

NUC. EN. MATUR., vol. 4, pt. 2, 1976

37994 NUCLEAR REACTOR CONTROL WITH REFLECTOR AND ABSORBER MEANS. Magladry, Robert E. (to Teledyne, Inc.). U. S. Patent 3,660,228. 2 May 1972. Filed 6 Nov 1967.

A nuclear reactor of the metal hydride control type is described. Metal hydride is in two portions, one in the core and the other external to the core, and means are provided for permitting flow of hydrogen between the two portions. Control of reactivity is obtained by providing heating means in one or both portions, the heating being responsive to a neutron sensor. (auth)

NSA, vol. 26, 1972

8919 BORON CARBIDE AGGLOMERATE. Maire, J.; Slonina, J.-P.; Sherer, A. (to Le Carbone Lorraine). French Patent 1,568,883. 21 Apr 1968. Filed 9 Feb 1968. (In French).

A method is described for the fabrication of agglomerates of boron carbide, B_4C, using a phenolic resin to bond the mixture of desired consistency. The various steps involve compression or machining in the crude form prior to polymerization of the resin, polymerization by heat, carbonization, and impregnation. The agglomerates of B_4C are used as neutron moderators. (P.C.H.)

NSA, vol. 25, 1971

38883 HIGH PURITY HOT PRESSED BORON NITRIDE. Mandorf, Victor Jr.; Montgomery, Lionel C. (to Union Carbide Corp.). U. S. Patent 3,660,027. 2 May 1972. Filed 6 May 1969.

Improved boron nitride articles having a unique acicular crystal structure, an oxygen content of less than 0.5 percent by weight, a density of at least 1.9 grams/cc, excellent hot strength, low and substantially isotropic coefficients of thermal expansion with no irreversible thermal expansion, excellent thermal shock resistance, moisture insensitivity, and improved dielectric properties are described. These articles are produced by treating conventional hot pressed boron nitride articles with a suitable solvent to lower their boron oxide (B_2O_3) content, and then sintering the treated material in an inert atmosphere at a temperature of from 1,600° to 2,100°C in the absence of pressure or mechanical restraint. (Official Gazette)

NSA, vol. 26, 1972

10057 WAPD-PWR-PMM-1806
Westinghouse Electric Corp. Bettis Atomic Power Div., Pittsburgh.
DEVELOPMENT OF BORON-BEARING MATERIALS FOR PRESSURIZED WATER REACTOR (PWR) CONTROL RODS. E. F. Losco and H. J. Snyder. Jan. 1958. 60p. $9.30(ph), $3.60(mf) OTS.

Boron-bearing materials were investigated for possible application as low-cost, high-performance PWR control rods. Two types of materials were considered. These included (1) stainless steel-enriched boron powder dispersion inserts contained in compartmented stainless steel structural alloys and (2) copper–natural boron carbide powder dispersions clad by roll-bonding to stainless steel. Both types of materials presented several serious manufacturing problems the solutions to which would result in little or no cost or performance advantage over the reference hafnium control rods. These findings, coupled with the very promising results obtained on silver–indium–cadmium alloys prompted continued development effort on the latter alloys at the expense of the boron-bearing materials. This report presents complete data and discussion of results obtained on the boron-bearing materials investigated. (auth)

NSA, vol. 13, 1959

38906 CORROSION OF STRUCTURAL MATERIALS IN THE BORON-CONTAINING SOLUTIONS USED FOR CONTROLLING THE POWER OF NUCLEAR REACTORS. Margulova, T. Kh.; Bursuk, L. M.; Bogatyreva, S. V.; Lipanina, A. A. (Moscow Power Inst.). Therm. Eng. (USSR) (Engl. Transl.); 17: No. 12, 13-17(1970).

Translated from Teploenergetika; 17: No. 12, 14-17(1970).

An investigation of corrosion of steel and Zr–2½Nb alloys in dilute boric acid solutions is described. It was found that the use of boric acid for control of reactors does not increase the corrosion rates for structural materials. When the solution is to be used in contact with steel, the solution should be made alkaline using KOH. Steel is recommended for use in fabrication of boric acid solution storage tanks. (J.R.D.)

NSA, vol 26, 1972

9. A Low-Cost Shim Safety Rod for Pool-Type Reactors*, *Robert D. Martin (U of Mich)*

A shim safety rod design for pool-type research reactors has been developed and tested at the Ford Nuclear Reactor (FNR) at the University of Michigan: The measured reactivity value of the new design, which uses cadmium as the neutron absorption material, is approximately 10% greater than that of the solid boron ($1\frac{1}{2}$ wt%)/stainless-steel design used at some pool reactor facilities. However, the total reactivity worth of the new design does not equal that of the common boron-carbide powder-filled oblong shell design.

The design, an assembly of eight $\frac{1}{2}$-in.-diam stainless-steel tubes arranged in a 2×4 array, was selected after a series of reactivity comparisons using 6-in.-long capsules of various cross sections. In each case, cadmium metal was used as the neutron absorbing media.

These preliminary evaluations indicated the tubular array design would have a reactivity worth of 96% of the boron-carbide powder-filled design. Based on these preliminary results, a full-sized shim safety rod was assembled.

This full-sized rod, when compared against spare boron-carbide and boron stainless-steel rods available at the FNR, exhibited a rod worth of 87% of the worth of the boron-carbide powder-filled design. The reason for this discrepancy between the capsule test and the full-sized rod comparison has not yet been resolved. As has been reported by other pool reactor operators using the solid boron stainless-steel design, the solid unit exhibits a worth of 80% of that of the boron-carbide powder design. The 80% value was confirmed by this reactivity comparison.

The comparison test was performed using a control-type fuel element located on the face of the reactor core. The positioning of the test rod was done using a manually operated lead-screw positioner. This location was chosen so that the test rod worths would not exceed the FNR license limitations (1.2% Δk) on experiment reactivity worth. The maximum reactivity worth noted in the comparison was 0.63% Δk.

One of the primary advantages of the tubular rod design over the boron-carbide and boron stainless-steel designs is that it can be fabricated at reactor facilities having modest machine-shop capabilities. No special forming or machining equipment is required. This simplified assembly was a primary design criterion of the evaluation program.

Fabrication of the full-sized comparison rod required less than $100 worth of materials and 60 man-hours of machinist labor. With experience and multiple-unit fabrication, a reduction in the required labor of more than 20% is expected.

A reactivity comparison has not been performed at a location interior to the reactor core. This is presently being considered, but because of experiment reactivity limits, that test will require an amendment of the FNR operating license.

*Sponsor: W. Kerr

ANS. TRANS., vol. 10, 1967

8915 PROCESS OF THE BINDING OF THE EXCESS REACTIVITY OF NUCLEAR REACTORS. Mattern, J. (to Licentia Patent-Verwaltungs GmbH). German(FRG) Patent 1,921,528. 23 Aug 1973. 3p. (In German).

The process works by means of movable control rods and on a certain position in the fuel element bundle individual adjustable rods of combustible absorber material. The share of excess reactivity bound by the combustible absorber rods is increased by reducing the share bound by the movable control rods in that certain adjusting positions are chosen for the combustible absorber rods. (GE)

NSA, vol. 30, 1974

22197 (RT/FI-(71)33) BURN-UP CALCULATIONS FOR LWR FUEL ELEMENTS CONTAINING Gd POISONED PINS. Mattucci, A.; Pistella, F. (Comitato Nazionale per l'Energia Nucleare, Rome (Italy)). 18 Oct 1971. 64p. Dep. NTIS (U. S. Sales Only).

Calculation procedures are presented to evaluate for LWR's: the cell constants of a Gd poisoned cell vs. irradiation (the codes developed are presented and the best choices for all the options available in the computation are suggested); and the reactivity values and power distributions in a fuel element containing Gd poisoned pins. It is shown that the procedures suggested can give more reliable information without enlarging the computation time requested, with respect to standard methods. 12 references. (auth)

NSA, vol. 26, 1972

15601 SCANNING ELECTRON MICROSCOPY OBSERVATIONS OF THE INTERACTION BETWEEN ADSORBED IMPURITY ATOMS AND CRYSTAL GROWTH OF GRAPHITE. Minkoff, I. (Faculty of Mechanical Engineering, Haifa, Israel); Nixon, W. C. pp 24-33 of Growth of Crystals. Vol. 8. /Sheftal, N. N. (ed.). New York; Plenum Publishing Corp. (1969).

From seventh international congress of crystal growth; Moscow (20 Jul 1966).

Observations were made with a scanning electron microscope of the interaction between impurity elements of the type that spheroidize graphite and graphite growth from the melt. The influence of the impurity element is to poison normal flake forms by adsorbing at the steps of screw dislocations. This leads to a progressive series of growth changes, the first of which is to form holes in the structure and promote branching of the crystal. Eventually, normal flake forms are completely eliminated and, as a final form, spherulitic crystals develop that may be considered as growth under conditions of melt impurity concentration, where complete coverage of the growing crystal is obtained. The undercooling at which growth takes place is that necessary to enable steps to traverse the crystal surface through the obtaining spacing of adsorbed impurity atoms. Observation of spherulitic graphite crystal surfaces shows that these are cellular in appearance, and growth hillocks are frequently capped by platelets. Factors that may influence growth on a spherical front are considered relative to growth processes incident on impurity adsorption. (auth)

NSA, vol. 26, 1972

8029 (CEA-N-1725) DEVELOPMENT OF METHODS FOR CALCULATING THE POISONS BURNABLE IN NATURAL WATER REACTORS. Mondot, J. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Div. d'Etude et de Developpement des Reacteurs). Jul 1974. 139p. (In French). Dep. NTIS (US Sales Only) $10.00.

Thesis.

A calculational method for reactor lattices containing self-shielded burnable poisons is presented. The calculation is divided in two steps. A transport calculation with a multicell model is performed using APOLLO (transport code with 99 energy groups and first collision probabilities). Some remarks are made on the multicell formalism used by this code. A method to determine the effective macroscopic constants of poison media is derived in order to obtain a good estimation of the poison worth with a diffusion calculation. Secondly, a two dimensional diffusion calculation with four energy groups is made either for the whole core of any desired part of it. A comparison between these calculations and results from critical experiments obtained by Babcock and Wilcox, shows good agreement regarding reactivity power distributions and various poisons worths. Some results of depletion calculations are also presented and an estimation of the influence of spatial non-uniform depletion of poison is given. (13 references) (auth)

NSA, vol. 32, 1975

6277 FUELS CONTAINING GADOLINIA. Murota, Kazuo: Miyamoto, Toshiki (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Japan). Toshiba Rebyu; 26: No. 8, 1019-23(Aug 1971). (In Japanese).

Almost all BWR's utilize initial and reload fuels containing gadolinia as the burnable poison. The reasons for using this kind of fuel, the nuclear and material characteristics of gadolinia, the design of the gadolinia-containing fuel, developmental research on the fuel, irradiation experiments, and the fabrication of the fuel are described. (auth) (Japan)

NSA, vol. 26, 1972

6. A Generalized Unit Cell Calculation for Lumped Poisons, *Barry N. Naft (W-NES)*

The inclusion of burnable poison and part-length control rods in recent nuclear designs, underscores a trend toward more sophisticated poison management. To add calculational support in this area, a specialized cell code has been automated. The automated system is currently available at WNES.[1] It is specifically designed to generate burnup-dependent cross sections, nuclide concentrations, and flux depression factors for lumped poisons. The guidelines for selection of such a model were that it:

1. provide a reasonable compromise between simplicity and a detailed nuclear model
2. be flexible enough to accommodate a wide range of geometries and material compositions
3. generate data that are consistent with multidimensional design codes
4. include burnup capability.

To accomplish these goals, the infinite lattice treatment in HAMMER[2] has been extensively modified. A homogeneous fuel buffer zone surrounding the poison cell has been included to provide a suitable boundary condition. An automated search matches the buffer-zone nuclear parameters to those of a fuel lattice cell. Also, a spatially dependent nuclide-depletion capability has been added. The HAMMER treatment has been internally coupled to an equivalent radial AIM[3] calculation, and a search has been programmed within the AIM to achieve consistent reaction rates. The new system contains a uniform and simplified input, plus a standardized data bank output tape for use in multidimensional codes.

The fuel buffer region is chosen to be sufficiently large so that an asymptotic fuel spectrum and flux boundary condition is imposed on the poison rod. It is a homogeneous composition of hydrogen, oxygen, and pseudoelements of pure scatterer, 1/v thermal absorber, and 1/v epithermal absorber. The buffer zone creates a well-behaved, yet realistic boundary condition on the poison cell. This enhances the "cosine current" approximation used in HAMMER, which assumes that the vector flux is proportional to the cosine of the scattering angle with the normal vector. This assumption conveniently allows a closed-form solution to the collision probability model for spatial flux distribution. It is weakest at irregular and anisotropic boundaries, but has shown an accuracy on a level between a P_3 and P_5 approximation, for isolated rods.[4]

A coupled space-energy solution is effected epithermally via the HAMLET-MUFT[2,5] treatment and thermally via THERMOS.[6] Space and energy should be coupled epithermally in light of the black or grey-black nature of poison rods. The approach also allows for the development of a detailed spatial nuclide-depletion calculation, which has been included. Previous techniques have depleted the poison rod homogeneously, or have only been concerned with thermal effects. The present approach considers the pointwise-dependent flux over the entire energy range, such that for a nuclide N with no precursors,

$$N(r,t) = N_0(r) \exp\left\{-\left[f_1 \sum_{g=1}^{54} \phi^g(r)\sigma_a^g + f_2 \sum_{h=1}^{30} \phi^h(r)\sigma_a^h\right] \Delta t\right\} ,$$

where

r = spatial point

g = MUFT fine-energy group

h = THERMOS fine-energy group

f_1, f_2 = fast and thermal flux normalization factors.

The thermal and epithermal flux are matched to the infinite fuel lattice spectrum at the buffer-zone outer boundary, and can be normalized to any power level.

The HAMMER cell model contains a fine-mesh transport solution, and as such is not entirely suitable for direct utilization in current multidimensional design codes. In an effort to provide a more consistent set of

(Cont'd pg 162)

nuclear parameters, an AIM calculation that performs a reaction rate search has been appended. The AIM contains an equivalent fine-mesh geometry and a double search is performed to match reaction rates while maintaining the neutron balance. Moving down in energy, $\Sigma_{absorption}$ for the poison pin is adjusted to match the pin to buffer ratio. Also, $\Sigma_{removal}$ in the buffer region is adjusted to match the ratio of absorptions to removals in the entire cell. The net result is a set of broad-group cross sections based on a fairly sophisticated transport model and forced to yield consistent rod worths and depletion rates in a diffusion theory calculation. Reduction to a more coarse-mesh or simplified geometry can be accomplished by matching reaction rates in a multidimensional diffusion theory code.

Cross sections generated by this method produce results that are in excellent agreement with experiment over a wide range of materials and geometries. Table I shows a calculation of several critical experiments.[7] Comparison with a "Reference k_{eff}" has been made to minimize calculational biases.

TABLE I

Comparison with Cold Clean Critical Measurements for Various Poison Rods

Experiment[a]	Calc k_{eff}	Ref[i] k_{eff}	Insertion Worth ($)
B_4C (powder)[b]	1.0015	1.0010	11.84
B_4C (powder)[c]	1.0003	1.0010	6.77
Ag-In-Cd[d]	1.0007	1.0020	6.52
Ag-In-Cd[e]	1.0017	1.0020	0.91
Pyrex glass (annulus)[f]	0.9997	1.0016	8.58
Pyrex glass (annulus)[g]	1.0006	1.0016	6.95
Pyrex glass (solid)[h]	1.0009	1.0016	5.33

[a]All SS304 clad, [b]69% TD, [c]63% TD, [d]0.335-in. o.d., [e]0.408-in. o.d., [f]0.228-in. i.d., 0.375-in. o.d., [g]0.316: i.d., 0.394-in. o.d., [h]0.396-in. o.d.
[i]Analytical calculation of unpoisoned core (ignoring support grids).

1. B. N. NAFT, "The HAMMER—AIM System," RD-ED-NA-75, WNES, Westinghouse Electric Corp. (March 1970).
2. J. E. SUICH and H. C. HONECK, "The HAMMER System," DP-1064, E. I. duPont de Nemours (January 1967).
3. H. P. FLATT and D. C. BALLER, "AIM-5, A Multigroup One Dimension Diffusion Equation Code," NAA-SR-4694, North American Aviation (March 1960).
4. A. MULLER and F. LINNARTZ, "On the Calculation of a Cylindrical Cell With Many Concentric Zones," *Nukleonic*, **5**, 1 (1963).
5. H. BOHL, Jr., E. M. GELBARD, and G. H. RYAN, "MUFT-4, Fast Neutron Spectrum Code," WAPD-TM-72, Westinghouse Electric Corp. (1957).
6. H. C. HONECK, "THERMOS, A Thermalization Transport Theory Code for Reactor Lattice Calculations," BNL-5826, Brookhaven National Lab. (1961).
7. A. VUKMIR and P. TOBIN, RE-X-M-521/531, WARD, Westinghouse Electric Corp. (1967).

ANS TRANS., vol. 14, 1971

37812 APPLICATION OF Gd AS A BURNABLE POISON FOR LIGHT WATER REACTORS. Orestano, F. V.; Pistella, F. (CNEN, Rome). Energ. Nucl. (Milan); 19: No. 5, 337-41 (May 1972).

The use of Gd as a burnable poison for light water reactors improves the reactivity evolution vs. burn-up as well as improves power distributions in the core. The calculational method developed to account for the effects connected with the depletion of Gd in actual operating conditions vs. lifetime is presented. Applications to a PWR as well as to a BWR are reported. 12 references. (auth)

NSA, vol. 26, 1972

1. A Review of Control-Rod Requirements for Water-Cooled Power Reactors / *T. J. Pashos (GE-San Jose), Invited Paper*

A reactor control system serves the following principal functions:

1) it provides a means of establishing and maintaining reactor operation at a desired power level by compensating for excess core reactivity allowed for fuel depletion, cold-to-hot core reactivity change, and buildup of neutron absorbing fission products;

2) it provides means of shutting down the reactor at an essentially instantaneous rate when called for by the safety system;

3) positioning of individual rods or banks of rods provides some degree of power shaping in large reactor cores.

All light-water-cooled and -moderated power reactors of current designs use vertically moving mechanical control rods dispersed throughout the core. Other than the Army Compact Reactors which use square box-shaped control rods, all water-cooled and -moderated power reactors use cruciform-shaped blades. In some reactors, the control rods are provided with fueled or nonfueled followers which fill the volume vacated as the absorber section is removed from the core.

The number of rods in a core is determined by the core size, amount of excess reactivity to be controlled, and the allowable strength of a single rod. Inasmuch as a large reactor core contains several critical masses, there is a close relationship between the rod size, worth, and spacing. Contrary to some beliefs early in the development of reactors, the most feasible control rod is not necessarily the one with the highest nuclear strength. Reactor safety requirements impose the "stuck-rod" criterion which provides that a control system must be capable of shutting down the reactor even though one rod is stuck in the fully withdrawn position. This then limits the size and worth of a rod and the spacing between rods for each reactor core.

Limiting the worth of a single control rod in turn limits the excess reactivity that can be incorporated to compensate for fuel burnup. If fuel lifetime beyond this limit is required, the mechanical control system can be augmented by the use of burnable poisons incorporated in the fuel, liquid poisons in the coolant, or removable core inserts containing absorber materials.

Many elements having high thermal-neutron-absorption cross sections have been considered for use in water-reactor control rods. However, evaluation over the years as to availability, fabricability, physics characteristics, metallurgical and chemical performance, and economics has reduced the number to relatively few elements such as boron, hafnium, europium, and a combination of silver, cadmium, and indium. These are used in specific physical systems: hafnium in its elemental form; boron as bulk B_4C in tubes or as an alloy with stainless steel; silver alloyed with cadmium and indium; and europium as an oxide dispersed in stainless steel.

The most economical and most widely used control rods are those made of B_4C in tubes or silver-cadmium-indium alloy. Hafnium and europium are expensive materials. Although their high material costs are offset somewhat by several neutron-absorbing isotopes in the case of hafnium, and a chain of absorbing daughter isotopes in the case of europium, the net costs of using these materials are indicated as being more expensive than using control rods containing B_4C or silver-cadmium-indium alloy[1].

1. WILLIAMSON, H. E., and F. H. MEGERTH, "Economic Evaluation of Control Rod Materials and Fabrication Processes," GEAP-4013 (May, 1962).

ANS TRANS., vol. 9, 1966

15609 PREPARATION OF METAL BORIDES. (to Reactor Centrum Nederland (Stitching). British Patent 1,257,544. 22 Dec 1971. Priority date 26 Mar 1968. Netherlands.

A method is described for the preparation of a metal boride, which comprises heating one or more halides or oxyhalides of one or more metals of the actinide or lanthanide or alkaline earth series of elements with B in the presence of one or more oxides of the corresponding metals(s), whereby a metal boride is produced. When halides or oxyhalides of the elements of the lanthanide or alkaline earth series are used, it is essential that the heating step take place in the presence of one or more oxides of the same element. The oxide is derived wholly or partly from a salt of the corresponding metal which is decomposed during the heating step to form the oxide. The metal borides produced are used as reactor control rods or as fissile material. (P.C.H.)

NSA, vol. 26, 1972

9263 MODERATOR AND CONTROL MATERIALS. Reactor Mater., 11: 220-5(Winter 1968-69).

BERYLLIUM — fabrication of tubes and ingot sheet from
BERYLLIUM ALLOYS — Be-Th, preparation of stable dispersions of
— Be-W, preparation of stable dispersions of
— preparation and properties of wire and rods from
BERYLLIUM OXIDES — rigidity modulus of, temperature and porosity dependence of
BORON CARBIDES — B_4C-Cu, thermal conductivity of dispersions of
BORON ISOTOPES B-10 — Al-^{10}B, thermal conductivity of dispersions of
CARBON — creep of glassy, stress and temperature effects on
CESIUM — sorption of liquid, by irradiated graphite
GRAPHITE — defect migration mechanism in irradiated
— sorptive properties of irradiated, for liquid cesium
HAFNIUM — mechanical properties of reactor-grade, effect of annealing on room-temperature
HAFNIUM ALLOYS — Hf-V, phase diagram for
— Cr-Hf, phase diagram for
HYDROGEN — reaction with binary alloys, hydride compositions from
REACTOR MODERATORS — properties of metal hydrides for, review of
— development of, review of
REACTORS — control materials development for, review of
— poison materials development for, review of (H.D.R.)

NSA, vol. 23, 1969

45990 MODERATOR AND CONTROL MATERIALS. Reactor Mater., 12: 89-94(Summer 1969).

Results of research and development are reviewed on graphite preparation and properties. Be metal and Be alloy metallurgy, BeO integrity, properties of Zr hydrides, properties of Nb hydrides, and properties of Ta hydrides. Results of investigations on nuclear poisons incorporating B and/or rare earths are also reviewed. (J.R.D.)

NSA, vol. 23, 1969

2930 MODERATOR AND CONTROL MATERIALS. Reactor Mater., 12: 236-47(Winter 1969-70).

Research efforts are summarized on graphite, Be, Be alloys, solid metallic hydrides, metallic control materials; and nonmetallic poisons. (J.R.D.)

NSA, vol. 24, 1970

3,625,821

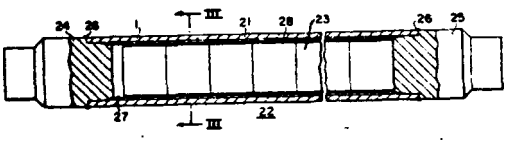
FUEL-ELEMENT COATING CONTAINING BURNABLE POISON

Herbert E. Ricks, Pittsburgh, Pa., assignor to Westinghouse Electric Corporation, Pittsburgh, Pa.

Continuation-in-part of application Ser. No. 499,407, Oct. 21, 1965, now abandoned. This application June 26, 1968, Ser. No. 740,218

Int. Cl. G21c 3/06

U.S. Cl. 176—68 **4 Claims**



A fuel element for a nuclear reactor having a fuel cladding tube with the inner surface of tube being coated with a retaining metal of low neutron capture cross section and having finely dispersed particles of a burnable poison disposed therein.

U.S. PATENTS, 1971

9842 (HEDL-SA-262) HOT PRESSING OF BORON CARBIDE IRRADIATION TEST SPECIMENS. Ross, W. A. (Hanford Engineering Development Lab., Richland, Wash.). 1 Oct 1971. Contract AT(45-1)-2170. 17p. (CONF-711030—4). Dep. NTIS.

From twenty-fourth Pacific Coast regional meeting of the American Ceramic Society; Anaheim, Calif. (30 Oct 1971).

A system is described for hot pressing boron carbide to any desired density greater than 65% theoretical. It was found that grain sizes can be modified by selection of starting particle-size and by annealing near the melting point. The B/C ratio can be lowered by addition of graphite powder. (J.R.D.)

NSA, vol. 26, 1972

48335 EMPIRICAL HELIUM RELEASE FUNCTION FROM THERMAL REACTOR IRRADIATED BORON CARBIDE. Russcher, G. E.; Pitner, A. L. (Westinghouse Hanford Co., Richland, Wash.). Nucl. Technol.; 16: No. 1, 208-15(Oct 1972).

Thirty-five sets of thermal reactor data were analyzed mathematically to derive a best fit function to predict gas release from boron carbide as a function of temperature, irradiation exposure, and material density. An exposure variable to account for difference in self-shielding in various reactor spectra was developed. The data used in the analysis included temperatures from 550 to 1200°F, irradiation exposures from 10×10^{20} to 31×10^{20} captures/g, and material densities of 2.0 and 2.5 g/cc (80 and 99% of the theoretical density). Within this range the function should predict gas release in all reactor spectra within the 20% estimated accuracy of the experimental data. Independent gas release data generated in fast and intermediate reactor spectra showed that the general form of the function is correct but that application to temperature conditions above the specified range may result in overestimates of gas release. (auth)

NSA, vol. 26, 1972

1550 (HEDL-TME-71-127) FUNCTION TO PREDICT LMFBR HELIUM RELEASE BASED ON BORON CARBIDE IRRADIATION DATA FROM THERMAL REACTORS. Russcher, G. E.; Pitner, A. L. (Hanford Engineering Development Lab., Richland, Wash.). 30 Sep 1971. Contract AT(45-1)-2170. 45p. Dep. NTIS.

An empirical function was developed to represent about 40 sets of irradiated boron carbide helium release data in the simplest form consistent with their observed radiation variables, and within their 10 to 20% accuracies. Helium release was chosen, rather than the customary gas release fraction, in order to use maximum data accuracy. Capture density was selected to relate helium release to the radiation exposure in that region of the specimen, where most of the helium was produced. Helium release data from natural boron carbide irradiated in the Hanford K thermal reactors were correlated with irradiation data from the EBR-II fast reactor in developing the helium release function. To demonstrate the applicability of this function to LMFBR problems, helium release is predicted for boron carbide irradiated in a neutron spectrum typical of Row 7 in the FTR. Within the range of the data used, helium release should be predictable for any fast or thermal reactor environment within an accuracy of about 20%. 9 references. (auth)

NSA, vol. 26, 1972

42566 OPERATING BEHAVIOR OF KWO FUEL ELEMENTS. Schenk, H.; Riedel, E. (Kernkraftwerk Obrigheim GmbH, Ger.). pp 562-5 of Reaktortagung, Bonn, 1971. Bonn; Deutsches Atomforum E. V. (1971). (In German).

From Reactor meeting; Bonn, Germany (30 Mar 1971). See CONF-710346.

A survey is given of the behavior of the KWO fuel elements. Behavior of the excess reactivity during the first and second operating cycle, arrangement of fuel elements in the first and second core, the decrease of boron concentration as a function of the medium burnup, axial burnup distribution, relative axial power density distribution, and activity behavior in the primary cooling agent are discussed. The inspection of the replaced fuel elements through the wet-sipping-test is described. (INIS)

NSA, vol. 26, 1972

9076 THE AUTOMATIC TIG (ARGON ARC) WELD FOR STAINLESS STEEL END CAPS IN THE POISON RODS. Schultema, D.; Meijer, F. A. Th. Lastechniek; 34: 123-5(May 1968). (In Dutch).

The welding current characteristics of the TIG weld are described for mechanically and electronically controlled current flow. The three phases of the welding operation—the start, the welding itself, and the cut-off—for stainless steel end caps in the Dodewaard Reactor poison rods are outlined in some detail. (J.S.R.)

NSA, vol. 24, 1970

51093 (LA-4164) FABRICATION OF BORON CARBIDE RINGS AND DISKS. Sheinberg, H.; Herrera, T.; Kostacopoulos, J. (Los Alamos Scientific Lab., N. Mex.). Apr. 25, 1969. Contract W-7405-eng-36. 11p. Dep. CFSTI.

Boron carbide components, varying in size from 3.2-in.-dia by 3.0-in.-long cylinders to 7.7-in.-OD by 6.2-in.-ID by 2.4-in.-long rings, were hot pressed to a minimum density of 95% of theoretical for use as neutron absorbers in a physics experiment. Powder characteristics, die design, pressing conditions, and the dimensional and density variations of the hot-pressed components are discussed. (auth)

NSA, vol. 24, 1970

59198 IMPROVEMENTS IN NEUTRON ABSORBERS. Shepherd, Leslie Robert; Hosegood, Samuel Brittan (to United Kingdom Atomic Energy Authority). British Patent 1,283,159. 26 Jul 1972. Filed 21 May 1970.

A thermal neutron absorber comprising a solid ceramic body incorporating silicon nitride is described for use in a high-temperature nuclear reactor. The absorber may be enclosed in a N_2 atmosphere within a gas-tight container. The function is to provide a "grey" control member. The ceramic body may be composed of a nitride or carbide of B, and reinforcing filaments of a refractory metal, such as Mo or W, may be incorporated. The absorber may be used as a control rod or for thermal neutron shielding. Details are included for preparation of the absorber material. (UK)

NSA, vol. 26, 1972

2. Performance of Silver-Indium-Cadmium Alloy Control Rods in Pressurized Water Reactors/*W. R. Smalley (WAPD), Invited Paper*

The first core of the Yankee-Rowe reactor contained 24 cruciform control rods, each fabricated from a solid solution alloy composed of 80% silver, 15% indium, 5% cadmium. This alloy has excellent nuclear control characteristics, including high thermal-neutron absorption, high resonance integral in the epithermal range, and long nuclear lifetime. In comparison with other control rod materials, it is economically attractive and easy to fabricate. The selection of the silver-indium-cadmium (Ag-In-Cd) alloy for the Yankee and other WAPD cores was based largely on previous Westinghouse development and testing[1,2].

For the Yankee Core I control rods, Ag-In-Cd cast billets were converted directly to cruciforms by extrusion. After machining, the rods were electroplated with 0.5-mil of nickel for corrosion protection in high temperature water. The plated control rods were then annealed at 600°C (1112°F) to promote a diffusion bond between the nickel plating and the silver alloy.

As part of the Yankee Core Evaluation Program, representative samples of nickel-plated Ag-In-Cd control rods from Yankee Core I were examined to determine the effects of reactor exposure on their irradiation stability. These samples were evaluated for changes in surface condition, dimensions, microstructure, and mechanical properties. The studies of the Ag-In-Cd control rods from Yankee included visual examination and destructive evaluation of material representing a wide range of neutron exposures, with a peak-integrated fast flux of about $4 \times 10^{20} n/cm^2 (> 1\ MeV)$, corresponding to ≈2300 effective full-power hours in the active core. The average irradiation temperature of the control rod alloy was estimated to be approximately 300°C (572°F).

Visual and metallographic examination after reactor operation revealed that large areas of the control rod absorbers were devoid of nickel plate and that the remaining plating had been attacked by the reactor coolant. Evaluation of the microstructures further indicated that the nickel plate and associated diffusion layer were not effective in preventing internal oxidation of the base metal.

The Ag-In-Cd alloy itself displayed good irradiation stability during reactor operation. Only small dimensional changes (up to ≈1% increase in volume) were observed; these changes were confirmed by similar decreases in density. Such changes are not considered significant to the performance of the Core I rods. The microstructure of the alloy was unchanged, with no evidence of new phases or significant precipitation. Similarly, the hardness and tensile properties of irradiated absorber samples showed no appreciable changes as result of exposure in Yankee Core I.

Evidence from the Yankee Core I follow programs showed that during reactor operation the nickel-plated Ag-In-Cd control rods performed satisfactorily despite any corrosion which may have occurred. The postirradiation evaluations, however, showed that the nickel plate did not prevent contact of the reactor coolant with the silver-alloy base metal. This, combined with visual evidence of rubbed areas and scratches on the control rods, could readily explain the release of radioactive silver into the coolant, as detected during the first refueling at the Yankee plant. As a result of these problems, the design of control rods for subsequent pressurized water reactors was changed to utilize extruded and drawn Ag-In-Cd bars inserted into stainless-steel tubes.

Stainless-steel-clad Ag-In-Cd rodded absorbers have performed successfully since September 1964 in the Saxton reactor. Control rods of this design were inserted at the time of the first head removal and have presented no problems to date. These rods have been tested at least twice a year for scramming characteristics and have behaved properly in all respects. In addition, stainless-steel-clad Ag-In-Cd control rods are being used in the SELNI reactor. Again, all evidence thus far indicates satisfactory performance.

1. COHEN, I., "Development and Properties of Silver-Base Alloys as Control Rod Materials for Pressurized Water Reactors," WAPD-214 (December, 1959).

2. WHYTE, D. D., and A. KRIEG, "Corrosion of Silver-Indium-Cadmium in Borated High Temperature Water," YAEC-147 (June, 1960).

ANS TRANS., vol. 9, 1966

29002 CONTROL RODS FOR WATER REACTORS. Synder, H. J. Nucl. Eng. Int.; 15: 350-2(Apr 1970).

A short survey is presented of the basic requirements of control rods for nuclear reactors dealing with function of the rod, its geometric forms, rod engineering requirements, methods of fabrication, irradiation effects, and economic influence. Rod properties and characteristics of control rod materials are emphasized. (auth)

NSA, vol. 24, 1970

15370 WAPD-A1W(NRM)-214
Westinghouse Electric Co. Bettis Plant, Pittsburgh.
LITERATURE SURVEY ON SELECTED ALLOYS CONTAINING METALLIC POISONS. A. B. Thomas, D. G. Freas, and E. S. Byron. June 1958. Decl. Mar. 24, 1959. 35p. Contract AT-11-1-GEN-14. $6.30(ph), $3.00(mf) OTS.

A literature search was made for information on metallic elements suitable for use as burnable poisons. The search was concerned primarily with availability, cost, alloy preparation and fabrication, phase diagrams, and corrosion resistance. The alloy systems are arranged in alphabetical order. (A.C.)

NSA, vol. 13, 1959

12376 REPLACEMENT HAFNIUM CONTROL RODS FOR THE BONUS REACTOR. Tolson, G. M. (Oak Ridge Gaseous Diffusion Plant, Tenn.); Davis, G. R. Nucl. Appl. Technol.; 8: 314-15(Mar 1970).

The selection of material and design, the preparation of a specification, and the manufacture of two replacement cruciform IIf control rods for the BONUS reactor are described. Difficulties in making and inspecting hafnium-to-Zircaloy welds are covered. A new type envelope gage for final dimensional inspection is also described. (auth)

NSA, vol. 24, 1970

14353 MEASUREMENT AND CONTROL OF PWR BOILER FOR THE FIRST PLANT OF MIHAMA POWER STATION. Tsukuda, Toshio; Inoue, Toshiaki (Kansai Electric Power Co. Inc., Osaka). Karyoku Hatsuden; 22: No. 5, 510-20(May 1971). (In Japanese).

The control principle of PWR type nuclear power stations is to control output of PWR boilers according to turbine output. To perform this, an automatic control system is provided in every system, and main systems are as follows. (1) Reactor control system. (2) Pressurizer control system. (3) Feed water control system. (4) Steam dump control system. Reactivity is controlled by position control of control rods jointly with boric acid density control in the primary coolant. A three element system is adopted for level control for the boilers and some functions peculiar to the PWR boilers. Response during automatic operation can meet to ±5%/min ramp change and to ±10% stepped change of rated load. The whole system also responds up to 50% stepped decrease of the rated load without causing trip, by combined use of the steam dump control system. In addition, the protection system, nuclear instrumentation, and process instrumentation are described. (Japan)

NSA, vol. 26, 1972

14852 (RDT-E-6-30-T(5-73)) ABSORBER PIN BORON CARBIDE PELLET. (USAEC Division of Reactor Research and Development, Washington, D.C.). May 1973. 14p. RSO.

The requirements for boron carbide pellets to be used in nuclear reactor control rod absorber pins are presented. (auth)

NSA, vol. 24, 1974

22408 (RDT-E-6-30-T(8-71)) ABSORBER PIN BORON CARBIDE PELLET. (Division of Reactor Development and Technology (AEC), Washington, D. C.). Aug 1971. 12p. RSO.

The requirements for boron carbide pellets to be used in nuclear reactor control rod absorber pins are presented. (auth)

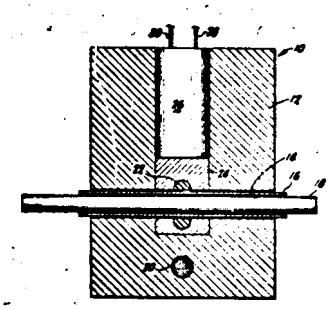
NSA, vol. 26, 1972

22407 (RDT-E-5-25-T(11-71)) CONTROL ROD ABSORBER PIN. (Division of Reactor Development and Technology (AEC), Washington, D. C.). Nov 1971. 15p. RSO.

The requirements for absorber pins for nuclear reactor control rod assemblies are presented. (auth)

NSA, vol. 26, 1972

3,728,544
METHOD AND APPARATUS FOR MEASUREMENT OF CONCENTRATION OF THERMAL NEUTRON ABSORBER CONTAINED IN NUCLEAR FUEL
Samuel Untermyer, Portola Valley, Calif., assignor to National Nuclear Corporation, Palo Alto, Calif.
Filed July 24, 1970, Ser. No. 58,039
Int. Cl. G01t 3/00
U.S. Cl. 250—83.1 **5 Claims**



Nuclear fuel samples containing burnable poisons are irradiated with epithermal neutrons and the neutron emission rate is counted to provide a measurement of fissile content through comparisons with the results of like steps performed on reference fuels. Where the fissile content is known, poison content is determined through irradiation of the sample with thermal neutrons. The presence of the poisons depresses the thermal neutron flux within the fuel and reduces the number of fission neutrons emitted. This emission rate is compared with the results of like steps performed on reference rods to indicate poison content. In another embodiment gamma emission by the fissile species is used in conjunction with thermal neutron interrogation to measure both fissile and absorber content.

U.S. PATENTS, 1973

32900 MATERIAL FOR THE PREPARATION OF NEUTRON ABSORBERS. Vobecky, M. Czech Patent 146,515. 15 Dec 1972. Filed date 21 Jan 1972.

The material consists of 0.05 to 95.00 wt% Cd powder and/or a mixture of Cd compounds and of 5 to 99.95 wt% plastics. The plastics may include substances such as polyethylene, polypropylene, polyisobutylene, polystyrene, polymethyl methacrylate, and polyethylene teraphthalate. A satisfactory density of target nuclei may be produced by the homogeneous dispersion of the powder target nuclide in the elemental form or in the form of a suitable compound or a mixture of compounds, in a plastic. If the target substance in the absorbing material is diluted using a filler, i.e., a plastic, the effect of self-shielding is suppressed in those nuclei which exhibit high neutron absorption values, e.g., Dy, Gd, Au, etc. The material also has good mechanical properties. (Nucl. Sci. Abstr. Czech.)

NSA, vol. 30, 1974

13051 BEHAVIOR OF UO_2–Gd_2O_3 FUEL. Wada, T.; Noro, K.; Tsukui, K. (Toshiba Research and Development Center, Japan). pp 63.1-63.3 of Nuclear Fuel Performance. London; British Nuclear Energy Society (1973).

From international conference on nuclear fuel performance; London, UK (15 Oct 1973). See CONF-731004-.

Some basic properties at high temperatures and in reactor behavior of UO_2-Gd_2O_3 fuel were studied. A single phase UO_2-Gd_2O_3 solid solution could be attained at relatively low temperature when co-precipitated powder was used. The difference of thermal expansion coefficient of UO_2 and UO_2-Gd_2O_3 solid solution in the composition range 0 to 8 wt % of Gd_2O_3 was negligibly small. No migration of Gd_2O_3 in UO_2-Gd_2O_3 pellets was observed when annealed at thermal gradients to 1,350°C and hot end temperatures to 2,100°C. In order to make clear the phase relationship, measurements were made in the composition range between 0 and 30 wt % Gd_2O_3. The fuel pin containing UO_2-3 wt % Gd_2O_3 pellets, irradiated in Japan Material Test Reactor (JMTR) with calculated maximum center temperature 2,280°C, revealed very different microstructures from those usually observed in irradiated UO_2 pellets. (6 references) (auth)

NSA, vol. 32, 1975

9075 INTRODUCTION TO PAPERS ON CONTROL RODS, FUEL ELEMENTS, FUEL ELEMENT STORAGE WELLS, AND PIPING SYSTEMS. Wassenaar, K. (Gemeenschappelijke Kernenergiecentrale Nederland, Arnhem). Lastechniek; 34: 118-22(May 1968). (In Dutch).

The Dodewaard Reactor core with its support construction and control rods are described and illustrated. The fuel element storage well and the high pressure steam cutoff valve are also considered. (J.S.R.)

NSA, vol. 24, 1970

31375 (BAW-3647-28) PHYSICS VERIFICATION PROGRAM. PART III, TASK 9. Quarterly Technical Report, July–September 1973. Webb, H. W.; Baldwin, M. N. (Babcock and Wilcox Co., Lynchburg, Va. (USA). Research and Development Div.). Feb 1974. Contract AT(11-1)-3109. 16p. Dep. NTIS $4.00.

In experimental Task 9, the fission rate in low-enriched UO_2 fuel pins and the neutron absorption rate in the boron of a simulated lumped burnable poison (LBP) material in a critical assembly are being measured. These rates are being compared for a few select core positions and for two types of LBP pins. (auth)

NSA, vol. 29, 1974

6. The Performance of B_4C Powder in the Dresden Control Rods, *H. E. Williamson (GE, San Jose).*

Small cracks were discovered in the vicinity of welds on the boron/stainless-steel alloy control blades in the Dresden Nuclear Power Station on February 18, 1961. Consequently, all the control blades were replaced with an improved blade then under investigation at APED. The improved control blade is formed by a single row of small diameter (~ 0.25 inch OD) boron-carbide-filled stainless steel tubes encased in a sheath.

To confirm the design, and to determine control-rod-lifetime limits, irradiation tests are continuing concurrent with operation. The tests include the irradiation and detailed destructive examination of special B_4C-filled stainless steel tubes representative of the tubes in the control blades. Representative tube samples were inserted in the instrument tubes of two fuel assemblies to provide experimental information on the effects of irradiation. Sample tubes have been examined, one after 2925 hours of operation (3.4 per cent B-10 burnup) and one after 11,040 hours of operation, (8.5 per cent B-10 burnup). Irradiation of two additional rods is continuing for examination at higher B-10 burnup.

The results of the examinations indicate that the performance of the B_4C powder has been satisfactory and well within design assumptions for the control rods.

Helium release from the boron carbide powder used in the Dresden control blades is higher than has been obtained in previous experiments[1,2,3] on hot-pressed boron carbide bodies as shown in Figure 1. This is most likely a result of additional helium release from recoil. Helium release from the Dresden boron carbide powder as a function of B-10 burnup is extrapolated from these measurements (obtained at less than 10 per cent B-10 burnup) and measurements (obtained at higher B-10 burnup in other experiments[1,2,3]) on hot-pressed carbide bodies as shown in Figure 1. Because of its extent, the extrapolation should be considered certain to only about a factor of two.

As a result of irradiation the boron carbide particles greater than 0.5 mm in diameter appear to be breaking down into two sizes: greater than 60 mesh, and greater than 140 mesh; there has, however, been no detectable increase in the < 140-mesh fractions. Measurement of the volume change in the boron carbide column indicates slightly increased compaction (0.7 per cent volume reduction) at up to 3.4 per cent B-10 burnup. Data are being obtained at higher B-10 burnup.

No evidence of significant chemical reaction between materials was observed.

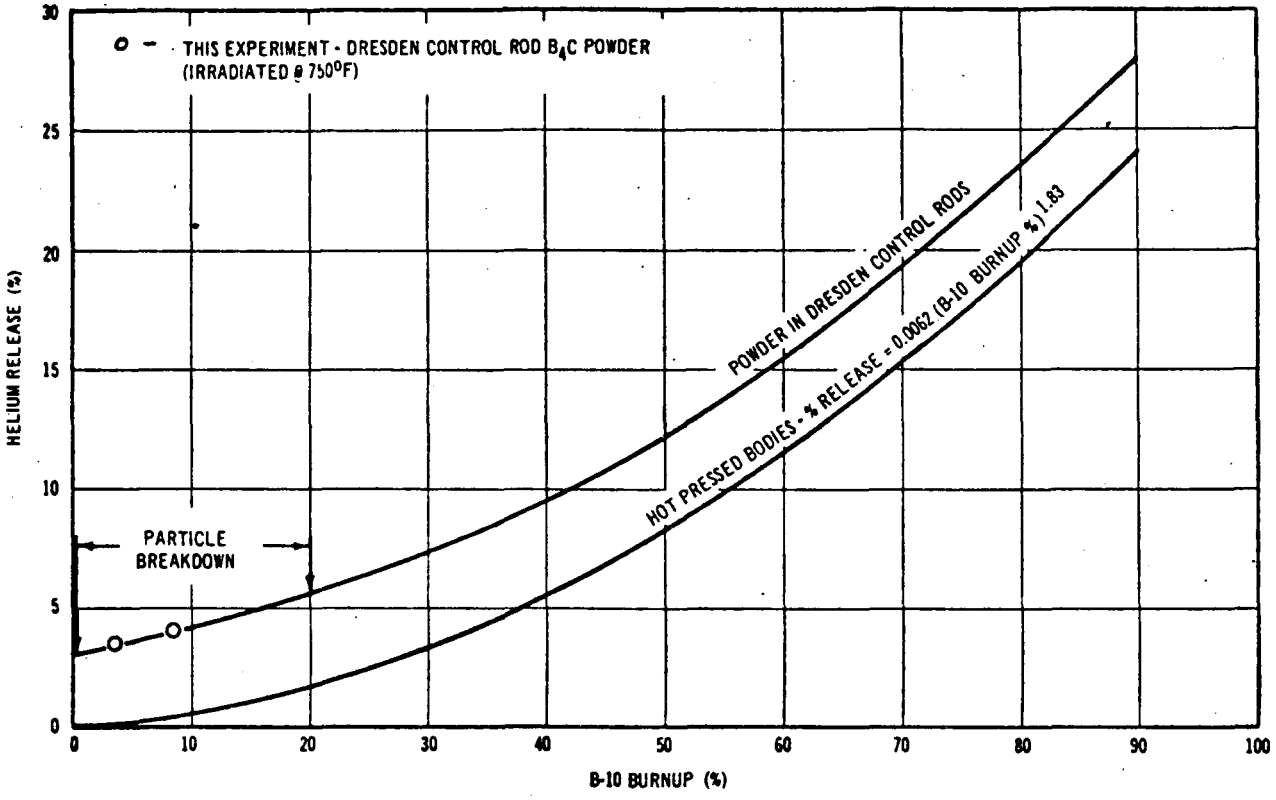
---

1. HOYT, E. W. and D. L. ZIMMERMAN, "Radiation Effects in Borides, Part I. Helium Release and Swelling in Irradiated Borides," GEAP-3743, (February 13, 1962).

2. HAMMAN, D. J. and P. SCHALL, "Radiation Effects on Boron Containing Compounds," BMI-1406, (January 6, 1960).

3. VALOVAGE, W. D., "Effect of Irradiation on Hot Pressed Boron Carbide," KAPL-1403, (November 15, 1955).

(Cont'd pg 168)



• Fig. 1. Effect of B-10 Burnup on Helium Release from B_4C at 750 F.

ANS TRANS., vol. 7, 1964

11866 (UCLA-12-849) CARRIER-FREE ISOLATION OF INDIUM FROM SILVER AND CADMIUM BY LIQUID–LIQUID EXTRACTION. Wood, R. A.; Wakakuwa, S. T.; MacDonald, N. S. (California Univ., Los Angeles). [1971]. Contract AT(04-1)-Gen-12. 15p. Dep. NTIS.

An analytical method was developed for the rapid and carrier-free separation of 2.81-d ^{111}In produced by the reactions $^{109}Ag(^4He,2n)^{111}In$ and $^{110}Cd(d,n)^{111}In$. The method is based upon the fact that indium in the presence of macro-concentrations of Ag(I) and Cd(II) preferentially extracts into n-heptane solutions of bis(2-ethylhexyl) hydrogen phosphate (HDEHP) from aqueous nitric acid media. The Ag(I) and Cd(II) remain quantitatively in the aqueous phase. The distribution ratios (K) vary from greater than 10^3 for In(III) to less than 10^{-2} for Ag(I) and Cd(II). The indium is quantitatively stripped from the solvent phase with HCl. The total separation time is less than 10 min. In the present investigation K was studied with respect to acid-solvent dependence, equilibration, and back extraction. Optimum conditions for maximum extraction were determined. (auth)

NSA, vol. 26, 1972

41392 BURNABLE POISON FOR NUCLEAR REACTOR. Yario, William R. (to Combustion Engineering, Inc.). U. S. Patent 3,663,218. 16 May 1972. Filed 12 Sep 1969.

An alloy is described for use as a burnable poison in nuclear reactors which consists of 0.5 to 14 wt % gadolinium and 0 to 4 wt % tin with the remainder being zirconium. (Official Gazette)

NSA, vol. 26, 1972

2. The Development of Boron-Aluminum Foils as Neutron Absorbers in Test Reactor Fuel Elements, *G. W. Gibson, M. J. Graber, M. F. Marchbanks, W. C. Francis (Id Nucl Corp)*

It is currently standard practice in high-power-density test reactors (e.g., ETR, ATR) to incorporate burnable neutron poisons in the fuel. Controlled additions and uniformity of dispersion of the poison (usually boron) have been serious problems to the fabricator as well as contributing to intracycle flux variations in the test reactor. When the fuel is made by conventional wrought metallurgical procedures, the boron is added to the molten uranium-aluminum prior to casting. Some variable quantity of the boron is lost through vaporization and a portion of the remainder tends to segregate on cooling. One obvious solution to this problem is through the use of powder metallurgy procedures to fabricate the dispersed fuel core. This technique is well proven and is currently specified for the above reactors but may result in some economic disadvantage. This paper describes a means developed for controlling the boron addition while still using the conventional wrought alloy technique to fabricate the fuel plates. This development has been adopted by a commercial fabricator to make ETR fuel elements.

In this approach, fuel plate cores of uranium-aluminum alloy are made by the alloying technique. Boron is incorporated in the fuel core as B_4C-Al foil made by powder metallurgy techniques. These foils are commercially available or can be readily made by compacting the well-blended powders of B_4C and aluminum and rolling into foils using a picture-frame technique. The foil can be located between the alloy core and the cladding or can be placed in the center between two half-cores. Both methods have been tested successfully in the MTR as sample platelets. The latter method has been tested in the ETR as full-scale fuel elements to exposures of 5×10^{20} fissions/cm^3.

Proper design of foil thickness and boron content limits swelling produced by the helium formation from the $^{10}B(n,\alpha)^7Li$ reaction to the same order of magnitude produced by the ^{235}U fission growth. Bonding of the sandwich layers and resistance to shear stress induced by bending have been adequate. Postirradiation blister anneal test results are equivalent to the standard alloy plates and only slightly less than powder metallurgically produced intermetallic fuels. Control of boron and uranium content is excellent.

Because recent bidding for ETR fuel elements has shown these foil elements to be economically attractive, and as a result of the successful irradiation tests, 260 elements are currently being procured for reactor use. In a carefully planned test program, these elements will be irradiated under progressively increasing severity of conditions to confirm the development work and to determine acceptability for unrestricted reactor use of the design.

ANS TRANS, vol. 11, 1968

Blank Page

VI. FUEL PIN WELDING

Blank Page

2. Tensile Properties of Electron-Beam-Welded 304 Stainless Steel,* *A. Arbel (WAPD)*

Type-304 stainless steel is used as a structural material in nuclear reactors. Tensile and deformation properties of cold-worked and electron-beam-welded 304 stainless steel were studied to show that the unique geometry of electron-beam welds facilitates high joint efficiencies. Electron-beam welds of high depth-to-width ratio were made on cold-rolled stainless-steel Type-304 at 90, 60, 45, and 30° to the rolling direction. Microhardness tests indicated that the tensile properties of the weld metal were similar to those of the annealed metal. Tensile specimens cut parallel to the rolling directions were slowly pulled to fracture. Deformation modes of the specimens were examined at the early stages of the loading by means of a grid etched onto their surfaces.

*Sponsor: W. R. Smalley.

Plastic deformation was first observed as a groove along the weld and adjoining heat-affected zone. Welded specimens had higher ultimate strengths than annealed specimens, although the weld metal had mechanical properties similar to those of the annealed metal. Analysis based on the mathematical theory of plasticity showed the high strength of the welded specimens was due to three-dimensional tension in the welds. These stresses resulted from the constraint imposed on the deformed soft-weld metal by the surrounding cold-worked base metal. In turn, this constraint resulted directly from the electron-beam weld's high depth-to-width ratio.

Specimens welded at 90° to the tensile axis always fractured along the weld, with the weld and heat-affected zone being strained plastically more than any other region. Longitudinal strains, shear strains, and rotation were observed in all specimens welded at 90° to the tensile axis. In these specimens, nonuniform plastic deformation spread into the base metal on both sides of the welds further than in the case of specimens welded at 90° to the tensile axis. This phenomenon resulted in greater elongations before fracture and slightly higher ultimate strengths than those of the butt-welded specimens.

In specimens welded at 60° to the tensile axis, fracture started along the weld but continued perpendicular to the weld, along planes of maximum shear stress. Specimens welded at 45 and 30° to the tensile axis exhibited the same stress-strain curves. Their UTS approached that of the cold-rolled unwelded specimens. Fracture of this type of welded specimen took place through the base metal across the weld. Table I summarizes the results.

TABLE I

The UTS, Uniform Plastic Elongation in the Base Metal and Nominal Strains to Fracture of 304 Stainless-Steel Specimens

Type of Specimen	Ultimate Strength (psi)	Uniform Plastic Elongation in the Base Metal (%)	Nominal Strain to Fracture in 2-in. Gage Length (%)
Annealed-welded	87 500	74	82.5
Cold rolled, unwelded	119 600	23	41.5
Cold rolled, butt welded	114 000	2.5	10.5
Cold rolled & welded, scarf angle = 60°	117 500	7	18.5
Cold rolled & welded, scarf angle = 45°	118 500	11.5	28.5
Cold rolled & welded, scarf angle = 30°	118 500	11.5	28.4

It is concluded that the unique geometry of the electron-beam weld (high depth-to-width ratio) and reasonable scarf angle (45°) resulted in very high joint efficiencies, while the UTS of the weld-annealed metal was well below that of the base metal.

ANS TRANS., vol. 11, 1968

4. Mechanism Causing Hydride Defects at End-Cap Welds of Zircaloy-Sheathed UO_2,* *A. S. Bain (AECL)*

Hydrogen reacts with Zircaloy to form a low-density zirconium hydride. With high hydride concentrations, the stresses from the density decrease can cause cracks in fuel-element sheathing. Cracked hydrided areas, observed with both powder[1-3] and sintered[4] UO_2, were attributed to an internal contaminant. To further study the behavior of Zircaloy-sheathed UO_2 which contained a hydrogenous contaminant, 12 elements with the following specifications were irradiated: sheathing—Zircaloy-2 or Zircaloy 4, 15.2-mm o.d. × 0.45-mm wall thickness; fuel—345 g of sintered UO_2 pellets (3.37 wt% ^{235}U in total U) with a density of 10.83 g/cm^3. Ten of the elements contained 20 or 40 mg H_2O or oil, or 100 mg of hydrated sodium silicate.

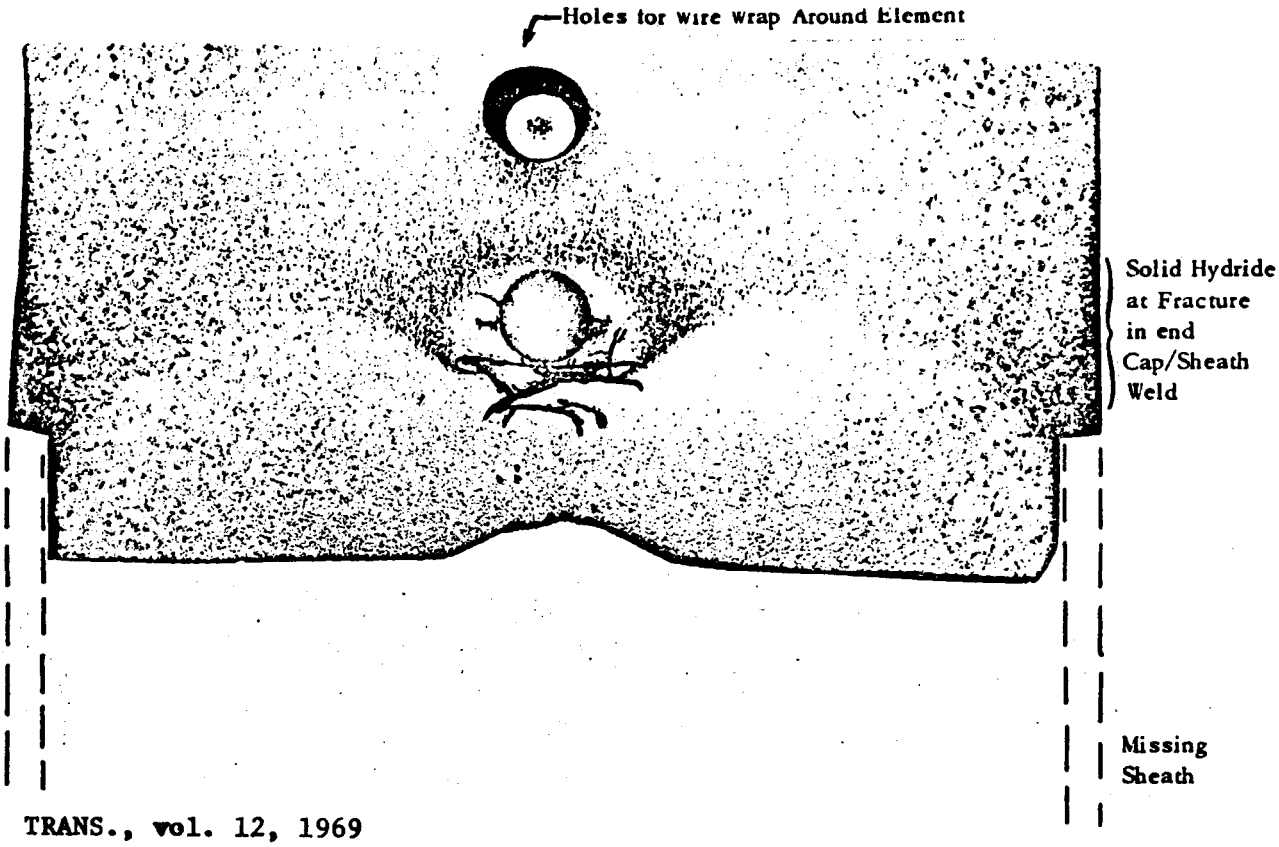
The elements remained intact for the 17-day irradiation at $\int \lambda d\theta$ = 55 W/cm and afterwards no blisters or other unusual features were observed on the outside of the elements. Hydrogen had reacted with the inside surface of the end cap, then migrated down the temperature gradient during irradiation. There was a sharp demarcation between low and high hydride areas, with the demarcation line corresponding to the 350°C isotherm (Fig. 1). A similar demarcation was predicted in a mathematical study[5] of the thermal diffusion of hydrogen in Zircaloy. At the end-cap surface, there were large cracks; presumably these occurred when the hydride concentration was high in that area, and the postirradiation observation of low hydride near the cracks is misleading.

*Sponsor: D. G. Hurst.

Cracks were also observed in the areas of high hydride concentration. With the design of end cap used, the cracked hydride areas were well away from the end-cap/sheath weld, and no defect occurred. With a different design, the hydride concentration could coincide with the weld and cause an end-cap defect (Fig. 2). Possibly other end-cap defects can be attributed to the same mechanism.[6]

These results emphasize the need to control hydrogenous contaminants in the fuel and illustrate the effects of different end-cap designs.

1. HW-76303 July-September, 1963
   HW-76304 October-December, 1963
   HW-81600 January-March, 1964
   HW-81601 April-June, 1964
   HW-81603 October-December, 1964
2. G. R. CASKEY, G. R. COLE, and W. G. HOLMES, "Failures of UO_2 Fuel Tubes by Internal Hydriding of Zircaloy-2 Sheaths," CEND-153, Vol. II, p. 77 (November 1962).
3. V. E. HAZEL, "Post-Irradiation Examination of Zircaloy-2 and Incoloy-800 Clad Fuel Rods Irradiated to 7000 MWd/TeU in the Consumers Big Rock Point Reactor," GEAP-5626, General Electric Co. (April 1968).
4. R. D. PAGE and A. D. LANE, "The Performance of Zirconium Alloy Clad UO_2 Fuel for Canadian Pressurized and Boiling Water Power Reactors," AECL-3068, Atomic Energy of Canada Ltd. (May 1968).
5. A. SAWATZKY and E. VOGT, "Mathematics of the Thermal Diffusion of Hydrogen in Zircaloy-2," AECL-1411, Atomic Energy of Canada Ltd. (October 1961).
6. M. G. BALFOUR, "Post-Irradiation Examination of CVTR Fuel Assemblies," WCAP-3850-1, 2, and 3, Westinghouse Electric Corp., Progress Reports for 1967 and 1968.



ANS TRANS., vol. 12, 1969

3,683,148

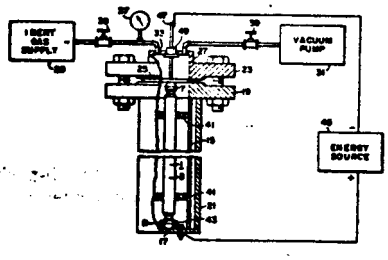
FABRICATION OF NUCLEAR FUEL ASSEMBLIES AND RESULTANT PRODUCT

Eugene S. Boyko, 720 Greenleaf Dr., Monroeville, Pa., and Joseph Campbell, 10135 Pearl Street, Penn Hills, Pittsburgh, Pa.

Continuation-in-part of Ser. No. 680,863, Nov. 6, 1967, abandoned. This application May 5, 1970, Ser. No. 33,156

Int. Cl. B23k 9/00

U.S. Cl. 219—137 8 Claims



A nuclear fuel assembly has an end plug and is sealed except for a small diameter axial bore in the plug. The plug provides a collar around the outer end of the bore which when fused provides sufficient material to seal the bore. Before the bore is sealed the assembly is filled with an inert-gas at high pressure. An electric arc is then established between an electrode and the collar to melt the collar sufficiently to seal the bore.

U.S. PATENTS, 1973

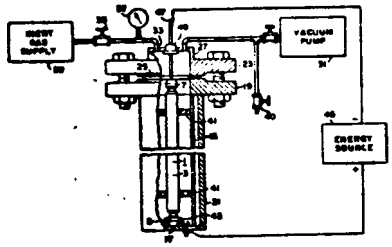
3,842,238

METHOD OF SEALING NUCLEAR FUEL ELEMENTS BY ELECTRIC WELDING

Eugene S. Boyko, Monroeville; Joseph Campbell, Pittsburgh, and Roger J. Wiggins, Allison Park, all of Pa., assignors to Westinghouse Electric Corporation, Pittsburgh, Pa.

Filed Dec. 8, 1970, Ser. No. 96,153
Int. Cl. B23k 9/00
U.S. Cl. 219—137 6 Claims

1. The method of constructing a pressurized nuclear fuel unit in which a predetermined quantity of nuclear fuel is inserted into a thin-wall elongated metal tube, said tube having a metal plug sealing one end of the metal tube and another metal plug having an axial opening sealing the other end of the metal tube to form a container having electroconductive walls, the axial opening constituting the only communicative path between the interior and exterior of the container and in which at least the metal plug having the axial opening is disposed within a sealed chamber which is thereafter pressurized



with an inert gas to establish a predetermined pressure within the container, the improvement of steps comprising, positioning a fusible welding electrode within the axial opening to make essentially line contact with the wall of the end plug having said axial opening, directing sufficient electrical current through said electrode, line contact and end plug to fuse the electrode by resistance heating, and continuing said fusion of the electrode by arc discharge welding until the axial opening is closed sealing the container under pressure.

U.S. PATENTS, 1974

8894 NUCLEAR REACTOR FUEL ELEMENT. Bresnick, S. D. (to United Nuclear Corp.). US Patent 3,804,710. 16 Apr 1974. Filed date 9 Dec 1970. 4p.

A tubular fuel element is described which has end plugs welded to each end with reproducible high quality welds. The tube contains fissionable fuel in any form but advantageously in the form of pellets partially filling the tube leaving a void space at one end for the reception of the fission gases. A compression spring of nickel, or an alloy of nickel, is located within this space to hold the fuel pellets in position. The end plugs, made of a zirconium alloy, are each provided with a head portion for engaging the fuel at the bottom end and the spring at the top. The surface of the head is spaced a suitable distance from the welded joint at the end of the tube and the head is connected to the inner portion of the end plug by a narrow neck portion. This construction prevents excessive stresses in the weld and adjacent tubing during service, and at the top end prevents the heat of welding from inducing the formation of a brittle nickel–zirconium eutectic in the adjacent materials. (Official Gazette)

NSA, vol. 30, 1974

26563 (HW-64889) UNBONDED CLOSURE WELDING PROCESS. Correy, Thomas B. Sr. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). 22 Apr 1960. Contract [AT(45-1)-1350]. 6p. Dep. NTIS.

Declassified 2 Nov 1971.

An unbonded closure with reactor-grade corrosion-resistant welds can be produced using a 0.060-inch thick Zircaloy-2 insert ring fusion welded by the tungsten inert gas process using argon shielding gas. The welds are made with direct current straight polarity in a small simple atmospheric pressure chamber that is continuously purged with argon gas. Approximately 200 pieces have been welded and given the high pressure and temperature autoclave test producing a lustrous black oxide coating of reactor grade. There has been no evidence of discoloration or corrosion products on the weld or in the weld zone. (auth)

NSA, vol. 26, 1972

9691 (INIS-mf-65) WELDING CORROSION TESTING AND ZIRCALOY-4 COMPONENTS. De Grande, A. (Comision Nacional de Energia Atomica, Buenos Aires (Argentina)). Nov 1970. 17p. (In Spanish). (CONF-701124–1). INIS.

From Metallurgical meeting; Cordoba, Argentina (Nov 1970).

The corrosion behavior in pressurized water and water vapor of welded Zircaloy-4 sheaths and closures for UO_2 fuel rods was studied. The following points were considered: effects produced by different surface treatments in short-duration autoclave tests, preparation of the Zircaloy samples with a nitrogen content between 200 and 600 ppM, and surface contamination with fluorine. From the tests made it was determined that in order to detect nitrogen in quantities between 250 and 300 ppM autoclave treatment of only one day in pressurized water at 300°C and 88 kg/cm^2 gives good results. In order to detect fluorine contamination a treatment of two days more in steam heated to 400°C at 20 kg/cm^2 pressure also gives good results. Increase of the pressure for the fluorine detection to 100 kg/cm^2 yields similar results. (tr-auth)

NSA, vol. 26, 1972

41397 PROBLEMS CONCERNING THE FABRICATION AND TESTING OF STRAIGHT-BEAD WELDED ZIRCALOY FUEL CLADS. Dressler, G.; Kleibeler, H.-J. (Metallgesellschaft AG, Frankfurt am Main. Vereinigte Deutsche Metallwerke AG, Duisburg, Ger.): pp 428-31 of Reaktortagung, Bonn, 1971. Bonn; Deutsches Atomforum E. V. (1971). (In German).

From Reactor meeting; Bonn, Germany (30 Mar 1971). See CONF-710346.

Tests for fuel clad fabrication via profiling of Zircaloy sheet to split tubes, straight-bead welding, and after-treatment for homogenization of the properties transverse to the tube cross section are described. (INIS)

NSA, vol. 26, 1972

32295 A DEVICE FOR ADJUSTING THE POSITION OF A WELDING ELECTRODE. Dufayet, J. P.; Marti, A. (to CEA, 75–Paris (France)). French Patent 2,199,492. 15 Sep 1972. 10p. (In French).

Description is given of a device for adjusting the position of a welding-electrode, in particular in register with a gap between two parts to be welded. That device is characterized in that it comprises a welding electrode located in a plane containing a probe constituted by an electromagnetic coil adapted to generate eddy-currents in the peripheral zone of the parts to be welded, an electronic unit adapted to control and measure these currents in the form of a signal permanently compared with a pre-determined threshold level, so as to detect the passage of the probe in register with the gap between the two parts, and a motor for moving the electrode and probe in parallel relationship with the parts. This device can be applied to welding an end cap to the fuel pin clad. (FR)

NSA, vol. 31, 1975

249144 Duncan, Robert; Barna, R.P. Westinghouse Electric Corp., Pittsburgh, Pa. (USA). **A fuel rod for nuclear reactors and a device for sealing said rod.** (In French). Barre de combustible pour reacteur nucleaire et appareil permettant le scellement de cette barre. **French patent document 2245055/A/.** Int. Cl. G21c3/40; G21c21/20. 18 Sep 1974. 10 p. Available from Institut National de la Propriete Industrielle, Paris (France); priority claim: 20 Sep 1973, USA.

A fuel rod for nuclear reactors, and a device for injecting pressurized gas into said rod and simultaneously closing the latter sealingly are described. The rod has an end-plug provided with a radial opening on the path of the circular weld connecting the plug to the rod-ends, said weld also obturating said pressurizing opening so that the fuel rod remains filled with pressurized gas at the moment of the welding operation in the device pressure-chamber. By means of such an arrangement, the fuel rod can be both obturated and pressurized in one operating step.

On decrit une barre de combustible pour reacteur nucleaire ainsi qu'un appareil permettant simultanement d'introduire du gaz sous pression dans cette barre et la fermer hermetiquement. La barre presente un bouchon terminal pourvu d'un orifice radial dispose sur le trajet de la soudure circulaire qui connecte le bouchon aux extremites de la barre, cette soudure obturant ainsi l'orifice de pressurisation de facon que la barre de combustible reste remplie de gaz sous pression au moment de la soudure dans la chambre de pression de l'appareil. Avec cet agencement, la barre de combustible peut etre obturee et mise sous pression en un stade.

CLOSURES: arc welding; HOLLOW FUEL RODS: closures.

Atomindex, vol. 7, 1976

38924 (WARD-4210-T-3-5, pp 2.1-19) CLADDING WELDABILITY STUDY. Elliot, D. N.; Ray, W. E. (Westinghouse Electric Corp., Madison, Pa. Advanced Reactors Div.). Jul 1971. In Topical reports.

A study was conducted to evaluate the effects of processing and fabrication variables on the weldability of Type 316 stainless steel fuel pin cladding. The program was limited to making test welds in different grades of Type 316 stainless steel cladding, with a constant set of weld parameters typical of those used in fuel pin end-closure welding. A total of 261 test welds were made, using semi-automatic tungsten inert gas welding; the test welds were examined using radiography, helium leak testing, metallography, visual examination, and dimensional inspection. The results obtained indicate that with 0.0015-inch end plug-tube interferences, the annealed material exhibits a 0.9% defect rate as compared with a 6.5% defect rate with 15 to 20% cold worked material. However, with no end plug-tube interference, the annealed material exhibits a 13% defect rate as compared to a 0% defect rate with 15 to 20% cold worked material. (auth)

NSA, vol. 26, 1972

1. Experimental Welding Developments for Power Reactors, *S. P. Grant (B&W)*

Welding plays an important role in efforts to improve power reactor designs. Three examples of experimental welding development work at Babcock & Wilcox are described in this paper.

Fuel-rod closures have been made with the plasma/electron beam process. Two approaches have been investigated. The first joins an end cap to the fuel tube in a distortion free manner without producing extraneous metal on the outside diameter which can interfere with rod insertion or removal unless an additional precise machining operation is performed. The second approach uses no end cap. An end closure of excellent shape is formed from the fuel tube itself, thus eliminating the necessity for fabrication of end caps.

An example of the first approach is shown in Fig. 1a. The welding operation took 10 sec in an atmosphere of 300 ppm helium. Helium was chosen because of its superior conductivity and low activation during reactor operation.

An example of the second approach is shown in Fig. 1b. Welds in Type-304 stainless steel and Zircaloy-4 were accomplished in this exploratory program.

Babcock & Wilcox is engaged in a cooperative program with the Naval Research Laboratory to develop and test a weld metal suitable for reactor vessels fabricated from A-543, Class 1, high strength steel. A-543 is a $3\frac{1}{2}\%$ nickel steel, and Class 1 stipulates an 85 000-psi minimum yield strength. B&W is developing the weld metal, and NRL is conducting the irradiation of impact and tensile specimens.

Welding of A-543 plate previously has resulted in a nil ductility temperature transition[1] in the deposited weld metal of 390 to 440°F when irradiated at a representative PWR operating temperature of 550°F to a fluence of 3×10^{19} n/cm^2 >1 MeV.

In this experiment, manual metallic arc welds were made with high and low copper and nickel contents, while phosphorus, sulfur, and vanadium were kept at 0.02% or less. Mechanical properties of all the welds as stress-relieved 40 h were found to meet A-543, Class 1 requirements. The fluence was 3×10^{19} n/cm^2 > 1 MeV.

With a copper content of 0.02%, a transition shift of 110° was obtained with high nickel (1.66%), whereas a shift of only 45° was found with low (0.71%) nickel. The shelves were 60 and 90 ft/lb, respectively.

With a copper content of 0.14% and low nickel (1.95%), a transition shift of only 110° to a final NDTT of +10°F was obtained. The as-irradiated shelf was 94°F.

These experimental results have shown that weld metal chemistry is a primary factor in NDTT shift during irradiation at PWR operating temperatures. One composition performed better than reference A-543 plate. The results indicate that copper content is a principle embrittlement factor. An apparent tolerance for minor copper additions was exhibited by one weld.

Additional experiments are underway to extend the investigation to submerged arc welds.

---

1. L.E. STEELE, J.R. HAWTHORNE, and C.Z. SERPAN, Jr., NRL Memorandum Report 1700 (May 15, 1966).

ANS TRANS., vol. 11, 1968

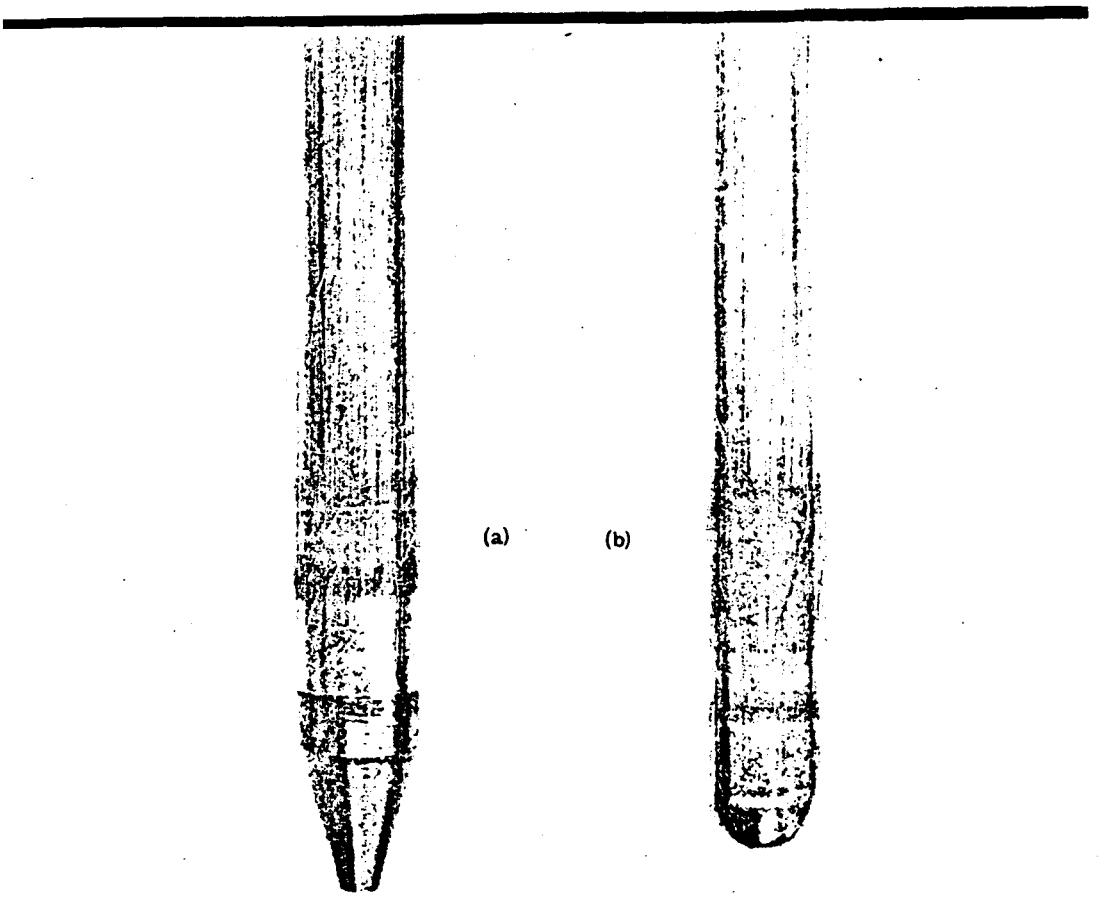


Fig. 1

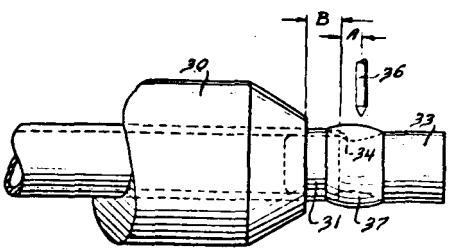
ANS TRANS, vol. 11, 1968

3,902,038

DISPLACED ELECTRODE PROCESS FOR WELDING

Lawrence J. Heichel, West Mifflin, Pa., assignor to The United States of America as represented by the United States Energy Research and Development Administration, Washington, D.C.

Continuation of Ser. No. 395,218, Sept. 7, 1973, abandoned.
This application July 22, 1974, Ser. No. 490,446
Int. Cl.[2] B23K *31/06*
U.S. Cl. 219—137 4 Claims



1. A method for butt-welding a relatively low-mass thin-wall tube to a relatively heavy-mass end block inserted in the tube, including the steps of:

a. positioning a welding electrode in a fixed relationship with the seam between said end block and said tube with said electrode being displaced from said seam toward said end block a distance within the range of ¾ to 2¼ times the wall thickness of said tube,

b. providing a chill block which completely surrounds said tube and which is in close contact therewith with said chill block being displaced from said seam away from said end block a distance within the range of 1.7 to 2.9 times the wall thickness of said tube, and

c. supplying a welding current to said electrode of a magnitude so that the top surface of the weld nugget extends down the tube substantially 1⅛ times the wall thickness of said tube.

AUGUST 26, 1975

U.S. PATENTS, 1975·

5703 WELDING IN NUCLEAR ENERGY. Nucl. Eng. Int.; 20: No. 224, 56-59(Jan 1975).

From 2nd international colloguy on welding in nuclear engineering; Duesseldorf, Ger. (23 Oct 1974).

A review is presented of the work reported at the second international colloguy on welding in nuclear engineering held at Düsseldorf in October 1974. One of the major topics was underclad cracking. Following some general comments on the subject, highlights are given from some of the papers. These fell into two distinct categories concerned either with the mechanism of underclad cracking, or with methods of avoiding underclad cracking and porosity. Some interesting methods for remote plugging of steam generator tubes were also discussed. (UK)

NSA, vol. 32, 1975

2468 METHOD OF ELECTRON-BEAM WELDING. Itoh, Fujio R. (to United States Atomic Energy Commission). U. S. Patent 3,602,685. 31 Aug 1971. Filed 28 Jun 1966.

A method of electron-beam welding is disclosed, particularly for welding alloys such as Zircaloy-2 or -4. Narrow, deep welds of minimal porosity are produced by defocusing the beam at its point of impingement on the joint. A short, narrow dike is provided on the surface of the joint within which pressure of the molten metal is built up. This pressure suppresses appreciable loss of alloying components from the weld metal. (Official Gazette)

NSA, vol. 26, 1972

48386 (HEDL-SA-329) FATIGUE-CRACK GROWTH IN TYPE 304 STAINLESS STEEL WELDMENTS AT ELEVATED TEMPERATURE. James, Lee A. (Westinghouse Hanford Co., Richland, Wash. (USA)). 3 Dec 1971. Contract AT(45-1)-2170. 22p. (CONF-720622-1). Dep. NTIS.

From Symposium on fatigue at elevated temperatures; Storrs, CT. (18 Jun 1972).

The propagation of fatigue-cracks in gas-tungsten-arc weldments of Type 304 stainless steel was studied at 75°F (24°C) and 1000°F (538°C). It was found that the crack growth rate for cracks propagating normal to the direction of welding decreased as the crack approached the weld, and then increased as the crack grew through the weld and out the other side. It is postulated that this is due to the residual stresses produced during the welding process, and that metallurgical variations in the weldment play a minor role. (auth)

NSA, vol. 26, 1972

257779 Kresslein, H.; Schmarch, E.U.; Zutz, L. Licentia Patent-Verwaltungs-G.m.b.H., Frankfurt am Main (F.R. Germany). **Light arc welding system for the welding of nuclear reactor fuel cans with end plugs.** (In German). Lichtbogenschweissvorrichtung zum Verschweissen von Kernreaktor-Brennelement-Huellrohren mit Endstopfen. **German (F.R.) patent document 1615367/B/.** Int. Cl. B23K 9-16. 18 Jun 1970. 7 p. 5 figs.

The arc welding device permits to carry out in series several welds or different kinds of welds when welding the end caps of cans for fuel rods of, e.g., superheater reactors. The welds can be carried out as circumferential seams, front seams, or longitudinal seams. To do this, the end of the can to be welded projects into a welding chamber filled with inert gas within which the welding torch is also arranged. The other end projects out of the chamber. By a torch-displacement system, the welding torch can be shifted along the longitudinal axis of the can and rotated with respect to this axis in at least one plane. In additon, a pick-up tube, which can be shifted along and rotated about the longitudinal axis of the can, is connected gas-tight with the welding chamber. It contains a stretching and centering device of the other end of the can. The welding chamber itself can be rotated about the longitudinal axis of the can. It is held in place by the stretching device of the can and a vacuum exhaust nozzle which are in alignment. (DG/PB).

FUEL CANS: arc welding.

Atomindex, vol. 7, 1976

3,460,237

METHOD OF MAKING A NUCLEAR FUEL ELEMENT

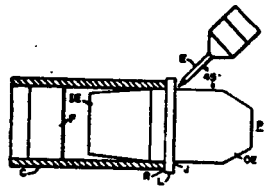
Donald R. McClintock, Irwin, Pa., assignor to Westinghouse Electric Corporation, Pittsburgh, Pa., a corporation of Pennsylvania

Continuation of application Ser. No. 275,884, Apr. 26, 1963. This application Dec. 20, 1966, Ser. No. 603,389

Int. Cl. B23k *9/00*

U.S. Cl. 29—474.3 **3 Claims**

A method is disclosed of joining a plug, having a land, to tubular cladding for nuclear fuel. The tubular cladding is very thin, about .010 inch, and may be composed of stainless steel. The joining is effected by fusion tungsten arc welding with the welding arc impinging at the junction of the land and outer extension of the plug.



U.S. PATENTS, 1969

259748 Murphy, J.L.; Turner, P.W. (Union Carbide Corp., Oak Ridge, TN). **Wire feeder and positioner for narrow groove electron beam welding.** *Weld. J. (N.Y.).* (Mar 1976). v. 55(3) p. 181-190.

The design criteria, wire feeder operation, and performance of automated electron beam welding apparatus are described. Information and discussions are included on wire feeder positioners, programming equipment, and computer control. Evaluations of the wire feeder in welding Al alloys, steels, and U are given.

ALUMINIUM: electron beam welding; STEELS: electron beam welding; URANIUM: electron beam welding; WELDING MACHINES: specifications.

Atomindex, vol. 7, 1976

260923 Podola, N.V.; Obolonsky, A.P.; Tsokol, E.L. AN Ukrainskoj SSR, Kiev. Inst. Ehlektrosvarki. **A method for measuring and stabilizing the diameter of the heating-spot on parts to be welded in an electron beam welding unit and a device for carrying out said method.** (In French). Procede de mesure et de stabilisation du diametre de la tache de chauffe sur les pieces a souder dans une installation de soudage par faisceau electronique, et dispositif pour la mise en oeuvre dudit procede. **French patent document 2255137/A/.** Int. Cl. B23k15/00. 19 Dec 1973. 16 p. Available from Institut National de la Propriete Industrielle, Paris (France).

The method described is of the type consisting in modifying the position of the focal plane of the welding electron beam issuing from an electron gun. The method is characterized in that during the welding operation, the welding electronic beam is deflected, by means of a linearly increasing pulse, so as to be directed towards an electron sensor specially designed for generating pulses proportional to the duration of the bombardment by electrons, said deflection being carried out in such a way that during the beam displacement, at least those electrons corresponding to one half of the beam cross section will impinge on said sensor, the output-pulse of which is of a duration proportional to the diameter of the electron beam. The invention can be applied to welding both small thickness parts and large thickness parts used in the nuclear field.

Le procede faisant l'objet de l'invention est du type consistant a changer la position du plan focal du faisceau electronique de soudage issu d'un canon a electrons. Il est caracterise en ce qu'on procede pendant le soudage, a l'aide d'une impulsion de croissance lineaire, a une deviation du faisceau electronique de soudage de maniere a le diriger sur un capteur de quantite d'electrons concu pour elaborer des impulsions proportionnelles a la duree de son bombardement par les electrons, la deviation etant executee de telle facon que, au cours du deplacement du faisceau, les electrons d'au moins la moitie de sa section transversale viennent frapper le capteur, dont l'impulsion de sortie a ainsi une duree proportionnelle au diametre du faisceau d'electrons.
L'invention peut etre appliquee au soudage de pieces de faible epaisseur aussi bien que celles ayant une grande epaisseur utilisees dans l'energie nucleaire.
ELECTRON BEAM WELDING: beam position: REACTOR COMPONENTS: electron beam welding.

Atomindex, vol. 7, 1976

44687 SPECIAL FABRICATION TECHNIQUES. **Reactor Mater.; 13: 92-105(Summer 1970).**

Results of recent innovations in reactor materials fabrication methods are reviewed. Information is included on powder metallurgy, melting, casting, welding, brazing, nonelectrolytic plating, electroplating, and explosive metal working. Development of nondestructive testing methods are also reviewed. Information is included on ultrasonic methods, infrared methods, and eddy current methods. (J.R.D.)

NSA, vol. 24, 1970

32286 CLOSURE OF FUEL CANS. **(to Societe Belge pour l'Industrie Nucleaire). Netherlands Patent 7,205,217. 18 Apr 1972. 7p. (In Dutch).**

The invention provides both an improved plug and a better welding process to seal off fuel rod cannings. (NL)

NSA, vol. 31, 1975

HIGH STRAIN CRACK GROWTH IN ZIRCALOY-2 WELDS

H. D. SOLOMON *General Electric, Research and Development Center P.O. Box 8, Schenectady, New York 12305*

The high strain crack growth rate in isotropic Zircaloy-2 welds was measured using strain cycling with controlled limits of plastic strain. The crack growth rate was found to be a function of the longitudinal plastic strain range ($\Delta\epsilon_p^L$) or the longitudinal elastic strain range ($\Delta\epsilon_e^L$) as given by the expressions

$$\frac{da}{dn} = 1.67\,(\Delta\epsilon_p^L)^{1.38}\,a$$

$$\frac{da}{dn} = 1.56 \times 10^{17}\,(\Delta\epsilon_e^L)^{9.17}\,a \quad .$$

This isotropic crack growth is compared to low cycle fatigue and crack propagation tests performed on the anisotropic starting material from which the welded specimens were fabricated. The behavior of the weldments was consistent with that observed in the anisotropic plates. The crack growth rate in the welds was between the upper and lower extremes measured with different orientations of the anisotropic plates.

NUC. TECH., vol. 29, 1976

260959 Tsutsumi, Masayori; Yoshino, Hiroyuki. Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan). **Method of securing end plugs to a nuclear fuel element.** (in Japanese). **Japanese patent document 1974-39115/B/.** Int. Cl. G21c3/10; B23k9/00. 17 Sep 1970. 3 p.

A method of securing an end plug to a fuel element employs a flanged end plug which is driven into the end of the tubular cladding member and welded by directing a torch toward the flanged portion at an inclined angle. Dimensions of the flange are such that the projecting edge thereof does not exceed the cladding in thickness, while the width of the flange is relatively short. Within a glove box, the end plug is driven into the end of the cladding until the lower face of the flange is flush therewith, thus accurately positioning the plug for welding. End plugs secured in accordance with the present method do not protrude beyond the outer diameter of the tubular cladding and thus neither plugs nor weldaments interfer with coolant flow or spacing means. (Owens, K. J.).

FLANGES: fuel cans; FUEL CANS: flanges.

Atomindex, vol. 7, 1976

29418 TELEVISION DEVICE FOR OBSERVATION OF A WELDING BATH. Vaugoyeau, H.; Schickel, G. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Service de Developpement et d'Essais d'Elements Combustibles). Metaux (Corros.-Ind.); 49: No. 587, 295-298 (Jul 1974). (In French).

From meeting on welding metallurgy; Marseille, France (8 Nov 1973).

The special conditions required for the welding of plugs on Pu fuel pins of fast neutron reactors have led to the development of a remote control observation process by television. This involves an original optical device based on the use of two converging optical systems: one with a long focal length giving little enlargement, the other with a short focal length to give a macrophotographic image. (FR)

NSA, vol. 31, 1975

17151 BIBLIOGRAPHY: ACOUSTIC EMISSION TESTING APPLIED TO WELDED JOINTS. Weld. Res. Int.; 5: No. 2, 78-85(1975).

The bibliography of publications on acoustic emission (stress wave) emission testing applied to welded joints includes 95 references, mainly covering the period 1969 to 1974. Seventeen of the references relate specifically to nuclear industry, covering topics such as the periodic inspection of steel pressure vessels for reactors, monitoring of reactor fuel pin end closure welds, and evaluation of radioactive waste capsule and welds. (UK)

NSA, vol. 33, 1976

31 (WANL-TMI-1765) LOAD TEST, GE MATERIAL EXTRUDED TUNGSTEN CUPS. (Westinghouse Electric Corp., Pittsburgh, Pa. (USA). Astronuclear Lab.). 7 Apr 1967. 2p. Dep. NTIS $4.00.

NSA, vol. 32, 1975

265878 White, D.; Woodacre, A.; Taylor, A.F. (Central Electricity Generating Board, Southampton, Eng.). **Advances in automatic welding control.**
Wyatt, L. M. (ed.). Welding research related to power plant. London. Central Electricity Generating Board. 1972. p. 574-591.

The development at the Reactor Fuel Element Laboratories, UKAEA Springfields, of a computer-based welding process control system, was aimed initially at the TIG welding of the end seals of nuclear fuel elements. The system provides for mixed multi-station operation with on-line real-time capability and can be used either as a research tool or for production requirements at competitive costs. The operation of the control system, the form of power source, and the servo motor control units are described. Typically, continuous or pulse-arc welding sequences can be digitally programmed on 0.1 sec increments, with current in 0.5 A increments up to a maximum of 256 A; up to three servo motors can be operated with speeds selected in 0.1 percent increments of their maximum. Up to six welding parameters can be monitored digitally at speeds from once every 10 msec. Some applications are described and it is shown that the equipment has wider uses outside the nuclear fuel element field. High quality industrial welding requirements can also be met and the system is not limited to the TIG process.

FUEL ELEMENTS: gas tungsten-arc welding; GAS TUNGSTEN-ARC WELDING: on-line control systems.

Atomindex, vol. 7, 1976

DISTRIBUTION

ERDA/FFTF-PO (3)

Director (2)
Assistant Director for Engineering Technology and Fuels

ERDA/Richland Operations Office (4)

Manager
Chief Patent Attorney
Chief, Nuclear Materials Management and Processing Branch (2)

E.I. DuPont de Nemours & Co.
Savannah River Plant
Aiken, S.C. 29801 (7)

J. T. Buckner
R. R. Herries
J. R. Hilley
R. T. Huntoon
D. J. Mahoney
W. R. McDonell
J. O. Scarbrough

HEDL (48)

M. J. Barr (2)
R. F. Boolen
W. F. Brown
T. T. Claudson
R. E. Dahl
E. A. Evans
E. W. Gerber
R. L. Gibby
E. M. Greybeck
E. N. Heck
R. J. Jackson
W. R. Jentzen
R. D. Leggett
D. E. Mahagin
R. B. McCord
B. J. Miller
R. C. Miller
M. K. Millhollen
D. H. Nyman
H. G. Powers
D. E. Rasmussen
H. C. F. Ripfel
J. D. Watrous
C. N. Wilson
R. E. Woodley
Central Files (20)
Publications Services (2)