AUTHORS AND PAPERS

The highly condensed summaries of papers and technical notes (below) are intended to assist the busy reader in determining the order in which to read the technical material. Biographical comments are for human interest.

FATIGUE AND BURST TESTS

Representative conditions of pressure cycling and geometry were used in this work to determine the bursting pressure and fatigue life of both unirradiated control specimens and sections of an irradiated in-pile stainless-steel pressure tube. Results indicate that irradiation increases the tensile strength and the low-cycle fatigue life of these tubes when subjected to pressure cycling.

Louis A. Waldman, shown seated, is a Fellow Engineer at Bettis Atomic Power Laboratory (BAPL) where he is currently associated with Reactor Development and Analysis. He obtained his BS in Chemical Engineering from the University of Toledo in 1951. Prior to his employment at Bettis in 1952, he attended the Oak Ridge School of Reactor Technology. He received his MS from the Carnegie Institute of Technology in 1956 and his PhD from the University of Pittsburgh in 1964, both in the field of Chemical Engineering. Menelaos Doumas is Manager of Irradiations Experiment Engineering at Bettis. Prior to his employment at Bettis in 1955, he was associated with the Central Research Laboratories of General Foods Corporation. He received his BS (Mechanical Engineering) from the College of the City of New York in 1947 and his MS in Mechanical Engineering from Columbia University in 1949.

RADIATION EFFECTS ON STAINLESS STEEL

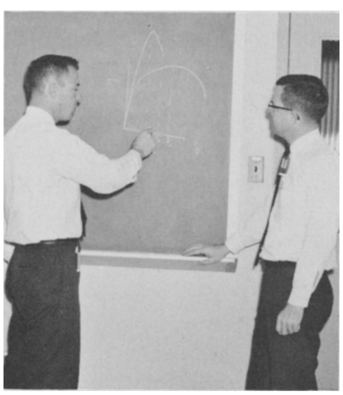
The mechanical properties and microstructures of Type-304 stainless steel were studied as a function of cold work, neutron irradiation, and testing temperature. Neutron irradiation increased the yeild strength and ultimate tensile strength of annealed and cold-worked specimens at 70°F and at 600°F. The increases in the strength and decreases in plastic stability produced by irradiation were combined by measuring the energy absorbed to plastic instability (area under the true-stress, true-strain curve up to the point of maximum load). This energy value was found to be an effective method for comparing the effects of the various variables. Cold work was found to produce large amounts of austenite-to-martensite transformation. Neutron irradiation was found to produce no measurable increase in martensite content.

The authors, from left to right, J. R. Low, Jr., Miss U. E. Wolff, and J. S. Armijo, are associated with the General Electric Company. Dr. Low (DSc, Carnegie Institute of Technology), an authority on the fracture of metals and the author of more than 24 technical publications, is a metallurgist with the General Electric Research Laboratory and a consultant to the Vallecitos Atomic Laboratory. Miss Wolff (MS, University of Goettingen) is a metallurgist-electron microscopist with the Vallecitos Atomic Laboratory; her area of specialization is electron microscopy and metallography of reactor materials. Mr. Armijo (MS, University of Arizona) is a metallurgist with the Vallecitos Atomic Laboratory; his special interests are in radiation damage and aqueous corrosion of stainless steel.

INHIBITING STRESS CORROSION CRACKING

Experimental results of treatments for inhibiting chloride stress corrosion cracking of austenitic stainless steels are presented. The inhibitors fulfill the theoretical requirements of a previously postulated mechanism of stress corrosion on crack induction.

C. R. Bergen is Supervisor of the Materials Development Group of the Allis-Chalmers Manufacturing Company Atomic Energy Division. Prior to joining Allis-Chalmers in 1962, he was with Alco Products, Inc. engaged in studying transport and deposition of radioactive corrosion products in pressurized-water reactors. He received his BS degree in Chemistry in 1950 and shortly thereafter joined the Colloid Chemistry Group at Continental Oil Company Research Laboratory. His work on surface chemistry, while there, provided background for the work reported in this paper.



DUCTILITY OF IRRADIATED STAINLESS STEEL

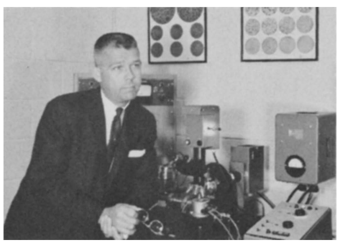
In general, the ductility of stainless steel, solution-annealed prior to irradiation, is better than steel, cold-worked prior to irradiation. One exception to this generality occurs under the conditions of irradiation followed by straining at 200°C, where low ductility is observed for irradiated stainless steel, solution-annealed prior to irradiation. Cold working followed by carbide precipitation at an intermediate temperature improves the ductility of irradiated stainless steel at 200°C.

Since 1959, W. R. Martin (shown on the right in the photograph) has engaged in research on effects of gaseous environments on the creep and short-time tensile properties of iron-base alloys as well as radiation effects on iron- and nickel-base alloys. J. R. Weir is at present the Supervisor of the Mechanical Properties Group in the Metals and Ceramics Division at the Oak Ridge National Laboratory. He has been associated with research on the mechanical properties of nickel-base alloys and radiation effects on alloys for reactor application.

CORROSION PRODUCT BEHAVIOR

Chemical and radiochemical analyses of fuel clad material, fuel cald corrosion product films, fuel clad deposits, and circulating corrosion products for the Yankee and Saxton nuclear power reactors are presented in this work. As would be expected, the specific activity of the metallic elements in both cases decreases in the order: clad, film, deposit, and circulating corrosion products. The pronounced decrease in activity from the clad to the film indicates that even the corrosion oxide film is largely deposited rather than originating from the local base metal. From the specific activity of the circulating insoluble corrosion products, it is difficult to explain quantitatively the observed radiation levels external to the reactor cores.

Paul Cohen, Manager, Chemical Development, Westinghouse Atomic Power Division, has over 15 years experience with Westinghouse in the area of water reactor coolant technology in naval and central station power application. His major assignments were in the development of the Nautilus, Shippingport PWR, and Westinghouse's current line of chemical shim reactors. G. R. Taylor, who manages the Chemistry Section reporting to Mr. Cohen, received his BS (Chemical Engineering, 1944) and DSc (Theoretical Chemistry, 1952) from Carnegie Institute of Technology. His ten years of experience at WAPD have involved homogeneous (slurry) reactor development and chemical-shim pressurized-water reactor development from its inception to its present operational status.



BORON STAINLESS CLADDING IN INDIAN POINT

In a program to confirm the use of boron stainless steel in the Indian Point reactor, manufacturing and welding techniques were developed for the production of high-quality tubing and suitable end-closure welds. Analyses showed that the boron in the steel had little effect on the steel's properties and that radiation effects on boron stainless steel do not detract from its utility as fuel cladding.

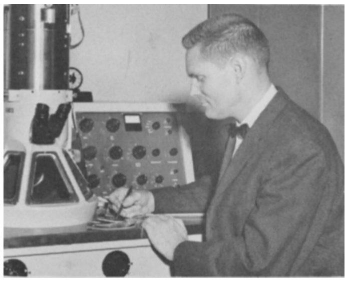
Charles R. Johnson has been active in the fields of materials development and applications since receiving his degree in metallurgical engineering from Purdue University in 1951. As a metallurgist with the Globe Steel Tubes Company (now Babcock and Wilcox, Tubular Products Division), he assisted in the development of methods to improve the quality of both stainless steel and low-alloy steel tubes. After a tour of duty in the Army, Johnson re-joined B&W as a materials engineer in the Atomic Energy Division. He was responsible for materials applications in the Indian Point Reactor and the N S Savannah *reactor in addition to supervising a group of metallurgists active in reactor materials research and development. When B&W built their Nuclear Development Center, Johnson was named Chief of the Materials and Processes Section. In this capacity, he is responsible for fuel fabrication development, reactor core materials research, and irradiation effects research on fuels and reactor core materials.*

TEST TO SIMULATE CRACKING

An iron chloride test was developed that closely simulates the intergranular attack on stressed nonsensitized Type-304 stainless-steel fuel cladding. This accelerated autoclave test can be used for screening prospective cladding materials and alloy modifications and developing a better understanding of the mechanism of attack.

A. E. Pickett, W. L. Pearl, and M. C. Rowland, from left to right, of General Electric Company, Atomic Power Equipment Department, are shown discussing the autoclave facility in which the intergranular attack studies were made. Dr. Pearl, Manager, Materials and Coolant Environment Unit, Vallecitos Atomic Laboratory, directs the applied development for establishing chemical, physical, and mechanical properties of materials associated with fuel cladding and construction and operation of nuclear power plants. Messrs. Pickett and Rowland are metallurgists in the Coolant Environment Unit and are involved in corrosion studies on boiling-water reactor fuel cladding and structural materials. Rowland is also involved in liquid-metal corrosion studies.

IRRADIATED YANKEE FUEL CLADDING



Type-348 stainless-steel fuel cladding removed from Yankee Cores I and II was evaluated to confirm its acceptability and to determine its irradiation stability. Visual and metallographic examination revealed that the surface condition of the irradiated fuel rods was excellent, with no evidence of corrosion, cracks, or other cladding defects. The microstructure of the cladding was essentially unchanged as a result of reactor exposure. Although the cladding showed marked increases in hardness and strength, the remaining ductility was demonstrated to be adequate.

W. R. Smalley has been with the Westinghouse Atomic Power Division since 1960. He is presently a Fellow Engineer in the Fuel Technology section, where he is responsible for development and evaluation of core materials for commercial power reactors. He has participated in irradiation studies for the CVTR pressure tubes, selection of materials for various pressurized-water reactors, and postirradiation examination of Yankee core components.

THIN-WALLED STAINLESS-STEEL-CLAD FUEL

Ten thin-walled Type-304 stainless-steel-clad fuel rods were irradiated as part of the Saxton core to a maximum burnup of 8900 MWd/t at a maximum heat flux of 410 000 Btu/(h ft^2). Postirradiation examinations revealed no evidence of failure or crack formation although the thin-walled clad was subjected to plastic deformation. The experiment demonstrated satisfactory performance of stainless steel in a borated PWR environment.

E. Paxson, D. R. McClintock, and H. M. Ferrari, shown left to right, are members of the Reactor Engineering and Materials Department, Westinghouse Atomic Power Division. Mr. McClintock is a senior engineer responsible for the Large Reactor Development Project Materials Irradiations Program. He joined Westinghouse in 1955 and was associated with the Bettis Laboratories and Westinghouse Testing Reactor prior to moving to APD in 1961. Mr. Paxson, who joined the Westinghouse Atomic Power Division in 1959, serves as manager of the Mechanical Analyses Group and is responsible for fuel rod design. Dr. Ferrari joined the Westinghouse Atomic Power Division in 1958 after graduate work at the University of Michigan. He presently serves as manager of the Fuel Technology Group.

ORDERED PACKED BEDS

In this study, quantitatively reproducible ordered packed beds were obtained consistently when spheres were dropped randomly into rigid rectangular columns. Irregular spheres as well as ball mixtures of two sizes with diametral differences as great as 5% in 10-to-50% mixtures could be packed in an ordered fashion. The beds can be fluidized and subsequently re-settled into an ordered array again. These ordered beds were found to possess great structural flexibility because they move in spring-like fashion.

Warren E. Winsche is Chairman of the Nuclear Engineering Department at Brookhaven National Laboratory under whose direction the ordered-bed studies were conducted. He has had extensive experience in the nuclear field including responsibilities for the design and construction of the Brookhaven Graphite Research Reactor and development work in separations chemistry at Savannah River. Herbert Susskind is a chemical engineer and Walter Becker a technical specialist who have been at Brookhaven since 1950 and 1956, respectively. Their work has been primarily in the high-temperature materials and liquid-metals areas.

STAINLESS-STEEL CLADDING FAILURES

Failures of Type-304 stainless-steel fuel rod cladding have occurred during irradiation in the VBWR; they are attributed to stress-assisted intergranular corrosion attack. The failures were invariably intergranular in nature and occurred in cladding material that did not contain precipitates within the grain boundaries.

W. H. Arlt, T. J. Pashos, R. N. Duncan, and J. P. Hoffman, shown left to right, as well as H. E. Williamson, shown separately on right, are affiliated with the General Electric Company, Atomic Power Equipment Department. C. J. Baroch, shown at left, is currently with the Atomic Energy Division of the Babcock and Wilcox Company. T. J. Pashos, as Manager of the Fuels Development Unit at Vallecitos, directs the development and irradiation testing of fuel and cladding materials for use in future nuclear power plant cores. Messrs. Duncan, Arlt, and Hoffmann are engineers in the Fuels Development Unit, as was C. J. Baroch at the time the paper was prepared. H. E. Williamson is currently a member of Core and Fuels Engineering, involved in the design and performance evaluation of boiling-water reactor cores.

FORMATION OF CYANIDE IN LIQUID SODIUM

Methods commonly used for measuring carbon or nitrogen in sodium do not detect cyanide, the formation of which has been demonstrated in liquid sodium at temperatures of interest.

Everett W. Hobart, Jr., has served for the past nine years as Supervisor of Analytical Chemistry at the AEC's Connecticut Advanced Nuclear Engineering Laboratory (CANEL). Following the recent shutdown of the CANEL project, he accepted a new position as assistant to the Research Director at Ledoux & Company. Robert G. Bjork (shown at left) joined the CANEL staff in 1962, after receiving his MS from the University of Connecticut. His most recent work has been devoted to characterization of the forms taken on by interstitial impurities in alkali metals.