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MONTHLY REPORT SUMMARY
AUGUST 1964

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MONTHLY REPORT SUMMARY
AUGUST 1964

Reactor and Fuels

A second comparative irradiation test of fuel elements of N-fuel uranium composition and of elements containing 400 ppm iron and 800 ppm aluminum in the uranium was completed. Post-irradiation bulk density measurements show consistently less swelling in the latter case.

A failure occurred in one of two N-single tube fuel elements with fluted outer surface which were under irradiation in the ETR. The failure, which occurred at an exposure of approximately 2500 Mwd/ton, was adjacent to an end cap and in the vicinity of a marker tab which had been spot-welded onto the cladding surface.

Irradiation testing of fuel fouling detectors fueled with thorium-uranium alloy has been successfully completed. Four detector probes are now in use in N-Reactor.

The 17 uranium alloys to be used in experimental irradiation studies to evaluate the effects of alloying and heat treatment on swelling have been fabricated into Zr-2 clad rods from which test specimens are being prepared.

All target elements for N-Reactor production test NR-8 have been delivered to N-Department. This delivery included 650 full length elements and 135 short target elements. Complete dimensional measurements were made on 108 of the enriched uranium driver tubes to be used in this production test.

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Visual examination of a series of Li-containing ceramic test elements after irradiation testing revealed no evidence of irradiation-induced changes. These elements were irradiated for sufficient time to give calculated gas contents of the test samples (GVR) up to 12.9.

Corrosion testing of defected Li-Al target elements in 280 C water has been completed. In those elements in which the target core was exposed directly to the coolant, swelling due to corrosion was observed within the first four hours and extensive corrosion damage and clad splitting had occurred at the end of the 297-hour test period.

Mechanical testing of closures from hot-headed, projection-welded and brazed elements have given closure strengths in excess of 30,000 psi. Highest strength was obtained from an element having six-ring projections.

Two transient heat transfer experiments were conducted to investigate fuel temperatures following the possible rupture of a main water supply line at N-Reactor. It was found that with a normal decrease in heat generation rate the maximum temperatures in the test section did not exceed 1100 F.

The time to boil water out of small production reactor process tubes filled with thoria fuel was calculated to be well over a minute for cases where water loss occurred after the reactor had been shut down for some time.

The crud film removed from a portion of the ruptured N-Reactor fuel element was less than 0.05 mil thick and would increase the surface temperature less than 1 C at a heat flux of 500,000 Btu/hr-ft^2.

No stress cracking was found on copper tubing and brass fittings exposed for five months to stagnant pH 9-10 ammoniated water and vapor at 55-150 C, the temperature and pH conditions proposed for the N-Reactor graphite cooling system.

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Corrosion evaluation in high temperature water of nickel plating techniques for aluminum and the effects of diffusion heat treatment indicate chemical nickel is superior to electroplates.

Coupons of Zircaloy-2 and 4 were exposed 58 days in-reactor and in ex-reactor control experiments to a 400 C (752 F) helium atmosphere containing CO, H_2, and water vapor. No evidence of acceleration of the corrosion rate in-reactor was noted.

The first stress rupture test of a section of an irradiated KER pressure tube has been completed. It ruptured in 492 hours at a temperature of 300 C (572 F) and a hoop stress of 61,000 psi. A similar specimen in the unirradiated condition would have been expected to fail at the same stress in 20 hours.

Under certain conditions hydrogen reduces the rate of oxidation of graphite-water vapor by factors of 3-1/2 and 7.

Continuous analysis of effluent gases has been demonstrated as a successful way of determining instantaneous rates of oxidation of graphite by water vapor.

The long term irradiation of N-Reactor graphite in the GETR is progressing satisfactorily.

Three fuel elements were delivered to the PRTR. A short core fuel element, a prototype for possible high power density PRTR testing, was prepared using UO_2-filled rods assembled into a 19-rod cluster.

Swaging and extrusion are being evaluated for fabrication of plutonium alloy wire for enrichment of uranium oxide pellet-filled rods.

A new swage rod loading hood was put into service which incorporates several improvements.

Excess oxygen in nonstoichiometric UO_2, introduced both by roasting and by adding U_3O_8, was found to equilibrate among the various UO_2 particle size fractions upon heating under vacuum at 1200 C for 80 minutes in a double Nupac can.

Examination after 106 hours of operation in the PRTR rupture loop revealed no fuel washout or changes in the appearance of an intentionally defected, pre-irradiated (240 Mwd/ton), swage-compacted UO_2 fuel element.

Failure in the modified end cap region of a PRTR rod, containing contaminated fuel material, indicates that end cap crevices are not the cause of previous fuel element failures.

Machining appears to enhance susceptibility of PRTR Zircaloy cladding to hydride attack.

The reaction layer on Zircaloy-2 capsules exposed to iodine at 400 C has been identified as alpha-zirconium.

The full length, Mark IX-B UO_2 inverted cluster fuel element was successfully irradiated in the PRTR rupture loop and displayed no visually detectable dimensional changes or surface fouling.

Capsule tests of EBWR UO_2-PuO_2 rods are performing satisfactorily under proposed 42 Mw EBWR conditions (60 w/cm^2 maximum heat flux) and to the exposure of most interest ($\sim$2.5 x 10^{20} fissions/cm^3).

Several dozen specimens of two overlapping Zircaloy strips, 3/16" wide by 0.010" thick, were spot welded together under water.

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Based on visual examination, the aluminum-clad fuel elements from C-1 Loop, which had been exposed in-reactor for 1400 hours to recirculating water at 260 C outlet temperature and pH 4.5 with phosphoric acid, did not appear to have nonuniform corrosion.

A 3/8" OD, 0.035" wall stainless steel tube, 18" long with a 0.110" tungsten heater wire down the middle, was vibrationally compacted with UO_2 to 85% theoretical density. This rod is to be used in fuel relocation studies.

Examination of a PuN-15 vol% Pu cermet irradiated to an estimated burnup of 35×10^{20} f/cm^3 showed that a portion of the specimen was converted to a fine powder.

The thermal expansion of a single crystal specimen of UO_2 was measured between 300 and 1600 C.

Irradiation was completed on the first of a series of UO_2 melting tests investigating the effect of stoichiometry on thermal conductivity during and after reactor startup.

A thoria element originally containing 680 ppm nitrate released 180.2 ml of gas, 170 ml of which was N_2, after 1.1×10^{20} nvt exposure in the MTR.

Five irradiated PRTR fuel rods with different fuel fabrication history were transferred to the Radiometallurgy Laboratory for sectioning. Approximately 12-inch rod sections will be slotted, capped, and subjected to simulated reactor coolant flow conditions to develop fuel washout data.

Enriched UO_2-15 wt% PuO_2, densified by pneumatic impaction to 95% TD, shows uniform plutonium distribution by autoradiography and chemical testing.

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No gross inhomogeneities in plutonium concentration were found in a sample of impacted, 64-hour ball-milled UO_2-20% PuO_2 analyzed by the electron microprobe.

Specifications were written for UO_2 powder to be used in the pneumatic impaction process.

Vibrational compaction loading of fused UO_2 particles in a thin wall Zr-2 clad, tube-in-tube fuel element resulted in greater than 86% theoretical density in both the inner tube and the outer tube.

All six of the small cans of BeO-nickel coated PuO_2 impacted material have been opened. The BeO matrix appeared to be bonded in all of the samples.

A highly loaded UO_2-niobium cermet, clad with thin wall niobium tubing and sealed with powdered niobium end caps was fabricated by pneumatic impaction.

The first use of rectangular tooling in the Dynapak machine was successful.

Previously reported differences in the tensile properties of Inconel X-750 have been shown to result from variations in grain size resulting from the conditions of heat treatment.

Oxidation rate data for Hastelloy X and Haynes 25 have been obtained at 1120 C (2048 F) in various pressures of oxygen, carbon dioxide, carbon monoxide, and atmospheric pressure air. Rate processes are generally parabolic.

Internal oxidation of cold worked TZM exposed 11 days to 1000 C (1832 F) 5×10^{-4} Torr air was observed; for annealed TZM, penetration was negligible.

Room temperature tensile tests on control specimens of various nickel base alloys exposed to 580 and 740 C (1076 and 1364 F) helium were completed and the data compared to that for specimens irradiated under similar thermal conditions to an exposure of 9×10^{19} nvt.

Preliminary results from high rate loading fracture experiments on A302B indicate a possible correlation between fracture phenomena and deformation behavior.

A 10% oxalic acid solution has been used to metallographically confirm the presence of martensite in cold-worked 304 stainless steel.

The special ETR C-7 loop test with a hydrogen addition to the coolant, which started in July 1964, with a varied charge of corrosion specimens, appears from water analysis to be progressing satisfactorily.

The first controlled pressure-temperature swelling capsule is operating successfully in a reactor at 450 C (842 F) and 1000 psi.

Examination of the replica from a U + Fe-Al specimen irradiated to 0.16 at% B.U. at 605 C (1121 F) showed small pores attached to a few of the second phase particles and almost no matrix porosity or tearing.

The radiation-induced reaction of carbon monoxide with water vapor was found to be very efficiently sensitized by helium. In gas mixtures containing up to 98% helium the yield of products based on total energy absorbed was invariant, with a G value of 15.

The oxidation rates of a highly purified graphite, SP-6, in air and in air with $C\text{-}Cl_2F_2$ inhibitor were found to be greater than rates for EGCR-type graphite. The results are anomalous since the SP-6 material contains a

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much lower content of vanadium thought to be the oxidation catalyst of importance.

The first boronated graphite samples recovered from the ETR irradiation facility co-roborated earlier results obtained in the Hanford reactors.

Increased stiffness in rolling Zr-2 strip has been observed on several occasions when a period of weeks has elapsed in the fabrication schedule, indicating a possible aging process. Laboratory tests show that increased strengths of 3000 to 6000 psi can be achieved by aging; however, these changes do not appear to be sufficient to cause the apparent increases in roll force which has been observed.

Irradiation of three tubular Zr-2 clad thorium-uranium fuel elements continued successfully in the ETR-P7 loop. Fuel swelling continues to be no more than expected from the solid fission products. The measured volume increase at 8500 Mwd/ton was 0.9%.

Irradiation of a uranium test rod containing a submicron dispersion of uranium carbide in the fuel has been completed. The test capsule has been shipped to Radiometallurgy for examination.

Techniques have been developed for strengthening the beta phase of unalloyed plutonium and for eliminating microcracks in high purity plutonium.

Molybdenum foils having three controlled carbon contents have been examined by x-ray diffraction after irradiation to 1×10^{20} nvt ($E > 1$ Mev).

Deformation studies in the electron microscope of irradiated molybdenum have demonstrated that moving dislocations interact with defect clusters and form jogs which impede their motion during glide.

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Large dislocation loops have been observed by transmission electron microscopy in specimens of irradiated carbon-doped molybdenum which were annealed at 750 C (1382 F) for two hours. These loops have been identified to be interstitial in character. The character of the smaller loops present in these specimens could not be established.

A discontinuity at approximately 110 C (230 F) in the differential thermal analyses heating curve of plutonium has been reported by investigators at the Mound Laboratories. The specific volume-temperature curve of high purity plutonium derived from heating rates of 1 C (34 F) per hour contained no discontinuity between 100 and 119 C (212 and 246 F). In the event of a change in either crystal structure or atomic structure in this temperature range, no volumetric change takes place.

The first shipment was made to Argonne of 810 EBWR UO_2-1.5 PuO_2 fuel rods.

A wet hydrogen, high temperature treatment was found to remove CO_2 and H_2O from UO_2 powder prepared from oxidation of the metal.

Fifty-five of 474 ultrasonically tested Zr-2 cladding tubes gave indications of defects in excess of 0.001 inch.

Two UO_2-62 wt% nichrome cermet discs, 0.6" diameter, totally clad in nichrome, were prepared by a Nupac technique.

Approximately 7500 20 wt% Pu-Al discs, 0.020 inch thick, have been fabricated to close tolerances by the recently developed spiral machining process.

Discs, containing 3 g/cc UO_2 (~69 wt% UO_2), were formed using an organic binder and carefully controlled heating rates. The 1.9" diameter by 3/8" thick discs machined satisfactorily and the material appears to be suitable for HTLTR control rod sleeves.

Two sample, 3 wt% Pu-Al, I&E fuel elements have been fabricated and non-destructively tested to determine core-to-clad contact area.

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