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OAK RIDGE NATIONAL LABORATORY

SPECIAL PROGRESS REPORT

PROGRAM 4400

MAY 1957

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C-85 Reactors - Aircraft
Nuclear Propulsion Systems

ORNL-2343

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OAK RIDGE NATIONAL LABORATORY
SPECIAL PROGRESS REPORT
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May 1957

Date Issued: JUN 10 1957

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C-85 - Reactors - Aircraft
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OAK RIDGE NATIONAL LABORATORY
SPECIAL PROGRESS REPORT

May 1957

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OAK RIDGE NATIONAL LABORATORY

SPECIAL PROGRESS REPORT

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OAK RIDGE NATIONAL LABORATORY

SPECIAL PROGRESS REPORT

May 1957

PROGRAM 4400 - REACTOR DEVELOPMENT

AIRCRAFT NUCLEAR PROPULSION PROJECT (AEC Activity 4401)

Aircraft Reactor Test (ART)

An experimental stress analysis of a full-scale aluminum mockup of the north head region of the ART pressure shell was completed under an ORNL subcontract at the University of Tennessee. The purpose of this study was to determine the stress concentration effects at the junctions of the pump barrels and the shell as a result of suspension and side loads exerted on the barrels by the support platform. The results of the test indicated that the design was adequate. Some minor modifications were required in the placement of instrument tube outlets.

Engineering Test Unit (ETU)

Work was started on assembly of the ETU reactor island parts. The final contouring on the upper beryllium hemisphere for the reflector-moderator region was completed. The first machining operation on the load ring forging was completed; the sodium outlet pipes were welded to the forging; and the strut and ring assembly is ready to weld to the forging.

The NaK furnace circuits are 75% complete. The two stationary assemblies for the NaK pumps and the main circuit electromagnetic flowmeters were welded into the systems. Design work was started on the main control panel layouts.

The cores for the two ETU radiators were welded and brazed and are being assembled into radiator units at the York Corporation. It is expected that the first ETU radiator will be completed by June 30, 1957 and the second by July 15, 1957. Work is under way on the fabrication of main and auxiliary radiator cores for the ART.

ART Facility Design and Construction

Package 3A contract work, which consisted of providing electrical distribution equipment for pipe and equipment heaters and a prefabricated building, including a dry-air plant and Government-furnished wound-rotor-motor controllers, was completed on May 10, 1957. A modification was made to the contract, however, at a cost increase of $17,480, and the

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contract time was extended 75 days. The new contract cost is $135,695, and the new completion date is July 24, 1957. The work under this modified contract will include the lube-oil fill and waste systems and portions of the louver hydraulic system.

Building and crane load tests were performed on the 30-ton crane to determine crane performance and building deflections at loads of 75,000, 80,000, and 85,000 lb, in anticipation of increased equipment weights. With loads positioned at critical points with regard to structural load-carrying ability, no vertical deflections were detected for any building column.

Developmental studies of disassembly equipment and methods were continued. Techniques are being developed for safely removing, disassembling, measuring, and examining the radioactive ART after operation.

Reactor Component Testing

A further test simulating fuel carryover in the off-gas system of the ART was made during water flow testing of the north-head mockup. With an expansion-tank air purge rate that roughly simulated or exceeded the maximum ART helium purge rate, the pumps operating at 2700 rpm and 645 gpm (ART design conditions), and the liquid level 1/2 in. above the expansion-tank floor, approximately 0.43 cm^3 of water was transported to the vapor trap in 6 hr. An extrapolation of this carryover rate gives an estimated rate of 0.10 $in.^3$ of fuel per day for the ART for continuous purging at the maximum rate.

The endurance test of a fuel pump operating with the fuel mixture $NaF-ZrF_4-UF_4$ (50-46-4 mole %, fuel 30) continued. A total of 3300 hr and 580 thermal cycles from 1100 to 1400°F has been accumulated. The primary NaK pump that was operating with NaK at 1200°F was shut down after 4400 hr of operation because the system pressure increased approximately 40 psi and the NaK level in the pump tank increased rapidly. The cause of this incident is being sought.

The first ART prototype NaK-to-air radiator was installed in THE Test Stand C. Room-temperature air flow measurements confirmed the design pressure drop characteristics reported by the vendor. High-temperature tests are to be initiated soon.

Radiator fin specimens with untreated edges that were cut by various methods and exposed to high-temperature (1200°F) flowing air at 1200°F in a test stand were examined. The loss of copper due to oxidation after 500 hr was less than 0.015 in., and thus the edge protection previously thought to be essential appears to be unnecessary.

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A prototype fuel dump valve modified to eliminate stem binding and misalignment successfully completed a 1500-hr test at 1300°F, including 50 openings and a 500-hr closure period. Leakage of the fuel mixture $NaF-ZrF_4-UF_4$ (50-46-4 mole %) was unmeasurable.

A prototype NaK dump valve and actuator assembly was received from Black, Sivalls & Bryson for testing. This valve has a guided plug. The plug and seat surfaces are Stellite 6.

An ART prototype vapor trap satisfactorily removed ZrF_4 vapor during a simulated reactor dump cycle in which 10 scfm of helium was passed through 1300°F fuel and through the trap for 5 min. Further tests at the design condition of continuous flow of 5000 liters per day through 1400°F fuel resulted in plugging at the trap inlet twice in 100 hr. Some redesign will be required to overcome low temperatures in the trap inlet pipe.

The prototype sodium vapor condenser plugged at an outlet fitting of reduced diameter after 1420 hr of satisfactory operation. Temperatures along the condenser were therefore reduced and made more uniform to ensure more complete removal of sodium before reaching the reduced-diameter region. The condenser has operated satisfactorily under the revised conditions for 720 hr. The NaK pump vapor condenser has operated satisfactorily for 860 hr at pump purge conditions.

In-Pile Experimentation

Disassembly work on in-pile loop No. 6 has revealed that fuel had moved along the pump shaft as far back as the copper cooling block and had solidified. The erratic pump behavior during operation was apparently due to thermal expansion having caused rubbing between the surface of the solidified fuel and the rotating pump parts.

Design Physics

Gamma radiation dose rates were calculated for a number of points outside the ART cell and were found in some areas to be far above tolerance levels. Decisions with regard to supplying supplementary shielding or restricting the occupancy of the affected areas have not yet been made.

When the reactor is drained, pockets of fuel will remain in the thermal sleeves around the NaK lines where they pass through the pressure shell at the bottom of the reactor. These pockets of fuel will create very large dose levels around the reactor that would seriously limit accessibility for maintenance work. It is planned therefore to provide annular shields of a tungsten-copper mixture around the NaK pipes between

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the pressure shell and the manifolds. A shield thickness of 1 1/2 in. will be required for a dose level of 0.1 r/hr, from this source, at the surface of the water bag.

An analysis of shielding requirements for the vapor trap on the fill-and-drain tank showed that La^{140} is by far the most important source of gamma rays in the fill-and-drain tank off-gas system. The most important mechanisms involved are (1) the decay of Xe^{140} in the off-gas system, which results in the deposition of Cs^{140}, which subsequently leads to La^{140}, and (2) the decay of Xe^{140} in the helium atmosphere in the fill-and-drain tank during a dump and the subsequent displacement of the daughter products into the off-gas system. Conservative calculations of the source strengths involved indicate a need for 2 in. of lead around the vapor trap on the fill-and-drain tank.

Power Plant Engineering

Various applications of nuclear-powered aircraft were examined in a parametric study, and the performance in each instance was compared with that of corresponding chemically fueled aircraft. The application in which the circulating-fuel reactor powered airplane appeared to be outstandingly superior to the chemically fueled airplane was the high-payload, subsonic airplane suitable for aircraft early warning, missile carrier, logistics, patrol, and other similar applications. The adaptability of the circulating-fuel reactor to unit and near unit shields makes it especially well-suited to these applications.

The testing of a standard 1/2-Mw radiator unit, fabricated by Pratt & Whitney Aircraft to ORNL specifications, was initiated at the NACA laboratories in Cleveland, Ohio. The test-rig equipment available at NACA will permit tests with a range of air velocities well beyond those attainable at ORNL. The test data should provide a firm basis for radiator performance estimates for high air velocity and low air density conditions.

Design studies of an apparatus for in-pile testing of moderator materials for use at high temperatures were continued. It was found that heat removal by thermal radiation and conduction through a 0.015- to 0.025-in.-thick gas film to a water-cooled surface appeared to be more promising than cooling by a high-velocity gas stream, as originally planned. A mixture of helium and argon can be used as the gas film, and the composition can be varied from pure helium to pure argon to effect a variation in heat load by a factor of about 4.

Heat Transfer and Physical Properties

Experimental studies of the heat transfer and fluid-friction characteristics of a delta-array heat exchanger were continued in the low laminar-flow region ($120 < N_{Re} < 500$). In this flow region the effects

of natural convection become apparent. With upward flow through the test unit, heat transfer was increased and pressure drop was decreased in comparison with the heat transfer and pressure drop obtained with downward flow.

A preliminary study to determine the pressure loads on the legs of adjacent channels of the ART fuel-to-NaK heat exchangers was completed. These channels serve in the reactor to prevent short-circuiting of the fuel along the direct path between the south and north headers and at the same time provide tube support. However, in defining the fluid paths the channels create a situation in which large pressure differences can exist across the channel legs. Since the clearance between the channel leg and the inner enclosing shell (shell IV) produces a high-resistance path for cross flow between channels and should reduce the pressure unbalances, the experimental study was made with and without such clearance. It was found that significant pressure differences (as much as 10% of the total pressure drop) existed across the channel legs without the clearance. The presence of the clearance tended to equalize the pressures on the opposite sides of the channel legs.

The enthalpies and heat capacities of $NaF-KF-LiF-UF_4$ (11.2-41-45.3-2.5 mole %, fuel 107) and $LiCl-CaCl_2$ (62-38 mole %) were determined in the solid and liquid states. The heat capacity of the fluoride mixture in the liquid state was found to be 0.40 Btu/lb·°F and that of the chloride mixture was 0.29 Btu/lb·°F.

The variable-gap method was used to determine the thermal conductivity of the mixture $NaNO_2-NaNO_3-KNO_3$ (40-7-53 wt %). The 0.4 Btu/hr·ft·°F value obtained agrees closely with earlier results obtained both at ORNL and at Battelle Memorial Institute. Similar studies of the LiCl-KCl eutectic gave a preliminary thermal conductivity value of 0.8 Btu/hr·ft·°F.

Metallurgy - Corrosion

Further screening tests of brazing alloys were conducted in seesaw-furnace apparatus with the specimens exposed to $NaF-KF-LiF-UF_4$ (11.2-41-45.3-2.5 mole %, fuel 107) for 100 hr at 1500°F. The newly developed Handy-Harmon brazing alloy designated Hi-Temp No. 91 (91.25% Ni-4.5% Si-2.9% B) was found to be slightly more corrosion resistant than the currently used Coast Metals brazing alloy No. 52 (89% Ni-5% Si-4% B-2% Fe).

A vanadium capsule exposed to fuel 107 for 100 hr in seesaw-furnace screening tests (1500°F hot zone, 1200°F cold zone) of container materials was found to have a heavy deposit of mass-transfer crystals in the cold zone. The capsule was protected from oxidation on the outside with a stainless steel jacket.

Thermal-convection loop tests of nickel-molybdenum base alloys have shown that, in 500 hr, exposure to fuel 107 results in attack to a depth

of more than 2 mils only in those alloys containing appreciable quantities (1 to 2 wt %) of aluminum, tungsten, or vanadium. Similar tests for a period of 1000 hr resulted in 1 to 2 mils of additional attack, that is, a total of 3 to 4 mils. Chromium in amounts up to 10% and niobium in amounts up to 5% did not affect the depth of attack, but the amounts of these elements increased as corrosion products in the fuel as the amount in the alloy increased.

Metallurgy - Fabrication

The effect of composition of the nickel-molybdenum base alloys on their mechanical properties is being investigated. Tests are under way on 18 alloys in the following system: 10 to 20% Mo, 4 to 10% Cr, 4 to 10% Fe, 0.06% C, 0.5% Al, 0.5% Mn, balance Ni. Thus far creep and stress-rupture characteristics have been studied at 1500°F and a stress of 10,000 psi in an argon atmosphere. The results indicate the following trends: (1) all the materials tested are superior to Inconel; (2) increasing the molybdenum content improves the creep properties; (3) for a given molybdenum and iron content, increasing the chromium content from 5 to 7% only slightly improves the creep properties; (4) there is no obvious correlation of rupture life with nickel content.

Metallurgy - Welding and Brazing

Tests were run to establish suitable procedures for welding and brazing the tube-to-tube sheet joints in the ART fill-and-drain tank heads. A low-restraint, trepanned type of joint is recommended as a means of minimizing the high degree of restraint exerted by the 1 1/2-in.-thick tube sheet. The feasibility of a brazing alloy preplacement procedure that incorporates annular sumps and three feeder holes to the joint was demonstrated.

Metallurgy - Nondestructive Testing

A new eddy-current instrument was designed for measuring thickness. This instrument, which is being constructed, will have both high sensitivity and high stability, and it will feature either variable-frequency operation or fixed-frequency operation with plug-in, transistorized oscillators.

A second metal identification meter was completed and is available for metal sorting or for conductivity measurements.

Resonance ultrasound was successfully applied to the detection of unbonded areas in Inconel-clad Cu-B_4C plates. Unbonded areas of less than 1/4 in. in diameter can be detected from either side of the plate, but the inspection is more reliable if the plate is inspected from both

sides. There was a 100% correlation with destructive tests of the several plates inspected.

Metallurgy - Mechanical Properties

The effects of environment, temperature, geometry, and cycle frequency are being explored in strain-cycling tests of Inconel. Preliminary results indicate that fine- and coarse-grained material behave similarly at 1300°F but that at 1500 and 1650°F fine-grained material shows definite superiority to coarse-grained material. Initial studies of the effect of geometry indicated that rod material was superior to tube material, but application of the same criterion of failure to both cases gave similar results for rod and for tube material. Thus far, changing the cycling period from 2 to 30 min has had no effect on the behavior of the material.

Radiation Damage

A forced-circulation in-pile loop fabricated of Inconel and filled with $NaF-ZrF_4-UF_4$ (60.4-27.4-11.8 mole %) was operated in a vertical hole in the LITR with the reactor operating for 235 hr, including 207 hr at full power. The total operating time for the loop was 332 hr. At full reactor power the maximum temperature of the fuel in the loop was 1600°F, the temperature drop in the fuel was 250°F, the Reynolds number was 3000, the maximum power density was 580 watts/cm^3, the dilution factor was 7.3, and the total power was 9000 watts.

The various components of the loop worked satisfactorily, except that after the first 100 hr the torque load on the pump motor began to increase, presumably because of radiation damage of the organic lubricant in the bearings. By the end of 125 hr of in-pile operation the pump turned more freely again, but there was intermittent chattering. Apparently the damaged grease had worked out of the bearing and it was operating dry. Operation of the loop was terminated by stalling of the pump, probably because of bearing seizure.

Chemistry - Fuel Research

The high-temperature crystalline phase α-$5NaF \cdot 2ZrF_4$ has a range of solid solution extending to 30 mole % ZrF_4. The temperatures of the inversion of α-$5NaF \cdot 2ZrF_4$ to β-$5NaF \cdot 2ZrF_4$ is lowered from 523 to 500°C by the solid solution formation. The phase β_2-$2NaF \cdot ZrF_4$, which melts incongruently at 544°C to α-$5NaF \cdot 2ZrF_4$ and liquid, undergoes an inversion to β_3-$2NaF \cdot ZrF_4$ at 533°C. The latter inverts to β_4-$2NaF \cdot ZrF_4$ at 505°C. Another modification, γ-$2NaF \cdot ZrF_4$, appears to exist below about 460°C, but its exact stability limits have not been determined.

Although a published partial diagram of the $LiF-YF_3$ system indicates a eutectic containing 18 mole % YF_3 at 744°C and no compounds of these materials, x-ray diffraction patterns of binary samples of $LiF-YF_3$ containing 20 and 25 mole % YF_3 show the presence of another phase that is, as yet, unidentified. X-ray diffraction patterns of $LiF-MgF_2-YF_3$ and $LiF-ZnF_2-YF_3$ also show the presence of the same unidentified material. It appears certain that the $LiF-YF_3$ diagram is more complex than the published information would suggest. Visual observations of the melting behavior of both the $LiF-YF_3$ samples indicate a solidus temperature of 690°C and liquidus temperatures 10 to 15°C higher.

Repeated experiments with the reaction

$$2UF_4 + M^o \rightleftharpoons MF_2 + 2UF_3$$

in $NaF-ZrF_4$ (53-47 mole %), where M^o is Cr^o or Fe^o, have indicated that the mole ratio of CrF_2/FeF_2 in equilibrium with the pure metals should be about 50. However, if FeF_2 and CrF_2 behave ideally in this melt, the reaction

$$FeF_2 + Cr^o \rightleftharpoons CrF_2 + Fe^o$$

should proceed to an equilibrium CrF_2/FeF_2 ratio of about 6×10^3. Recent attempts to establish equilibrium concentrations experimentally have shown that the reaction proceeds nearly to completion. Concentrations of 100 to 150 ppm Fe^{++} result from initial charges of 6000 ppm Fe and an excess of Cr^o. The data indicate clearly that the stoichiometry expressed in the simple equation is correct.

The study of the solubility of argon in $NaF-ZrF_4$ (53-47 mole %) was completed. The results show that Henry's law is followed and that the solubility increases with increasing temperature. As was the case in the related solvent $NaF-ZrF_4-UF_4$ (50-46-4 mole %), the solubility decreases with increasing gas atom size. The values found for the solubility of argon in this solvent are 0.506, 0.807, and 1.20×10^{-7} moles/cm^3 of melt/atm at 600, 700, and 800°C. The corresponding Henry's law constants for helium in the same solvent are 2.16, 2.92, and 4.20×10^{-7} at 600, 700, and 800°C.

Chemistry - Fuel and Coolant Production

The 250-lb fluoride purification facility processed 11 batches totaling 2750 lb of fluorides during May. It was necessary to terminate purification operations during the month, however, because of a shortage of receivers. Some 2000-lb of low-hafnium $ZrCl_4$ was converted to ZrF_4. Repeated difficulty with the dust filter on the conversion unit resulted in only 78% recovery of the end product, ZrF_4. The experimental-scale fluoride purification facility was again utilized chiefly for processing of 50-lb batches of $NaF-KF-LiF-UF_4$ (11.2-41-45.3-2.5 mole %).

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Shield Design

An experiment is being designed for qualitatively determining whether the gamma rays from the inelastic scattering of fast neutrons in air make an appreciable contribution to the gamma-ray dose rate in a crew compartment. Calculations indicate that at sea level the dose rate from gamma rays which are produced from slow-neutron captures in air is probably much larger than the dose rate from gamma rays produced by inelastic scattering. The relative importance of inelastic scattering should, however, increase with increased altitude.

Bulk Shielding Facility (BSF)

It became apparent that it would be necessary to operate the 10-in.-dia NaI(Tl) crystal with a well in it to avoid Compton backscattering, and therefore the shield-collimator assembly for the crystal was redesigned. Some consideration was given to the possibility of installing sources within the spectrometer tubes for calibration checks during operation, but no final plans were made.

A rig for testing the control plates for the proposed UO_2-stainless steel core for the BSF reactor was placed in operation and is now undergoing modifications to improve some features of performance. The data obtained thus far indicate that the time required to drop the plates is greater than the calculations indicated. With an initial 8-g acceleration (661-lb spring) 0.224 sec was required for the plates to drop 12 in. of a 15-in. stroke, whereas the calculated time was 0.15 sec. Both these times include magnet release time. For the free-falling rods presently used in the BSF reactor the drop time is 0.68 sec, and therefore the improved magnet and the accelerated plates do give a substantially shorter drop time. The dash-pot decelerator stopped the plates at the bottom of the travel with very little shock.

Lid Tank Shielding Facility (LTSF)

A new series of experiments with advanced shielding materials was begun. The configurations being tested are of particular interest to Pratt & Whitney Aircraft. This series of experiments differs from the preceding series primarily in that more beryllium is being used between the source plate and the heavy gamma-ray shield and the instruments are being run in borated water rather than in oil. The first material investigated was Hevimet. Fast-neutron dose rates, thermal-neutron fluxes, and gamma-ray dose rates were obtained behind configurations made up of 4 in. or more of beryllium, 4 in. of Hevimet, up to 36 in. of lithium hydride, and, in some cases, 1 in. of stainless steel. Experiments on other materials are in progress.

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Tower Shielding Facility (TSF)

A determination was made of the experimental errors that affect the reproducibility of measurements taken at the TSF. Three different geometrical configurations were used for 27 dose measurements, and the data were found to be reproducible to within a probable error of ±2%. Seven calorimetric power determinations made since March 1 were reproducible to within a probable error of 1.5%.

Tower Shielding Reactor-II (TSR-II)

Engineering drawings of the TSR-II fuel elements were completed and from them the metal-to-water ratio in the core region was calculated to be 0.573. A program was started to develop a method of fabricating the fuel elements. Aluminum alloy 6951 is to be used rather than 1100 (2S) because of its superior hardness characteristics. Heliarc tack welds were used to assemble the first dummy element, and the results were satisfactory. The over-all operating time of a prototype safety scram mechanism was measured to be about 60 msec.

Edited by A. W. Savolainen

Approved by Alvin M. Weinberg, Director

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