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MONTHLY PROGRESS REPORT

HEAVY WATER POWER REACTOR PROGRAM

OCTOBER 1963

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POWER REACTOR DEVELOPMENT ACTIVITIES
SAVANNAH RIVER OPERATIONS OFFICE
OCTOBER 1963

I. RESEARCH AND DEVELOPMENT

AEC/AECL COOPERATIVE PROGRAM

On-Power Refueling Machine for Canadian Deuterium Uranium Reactor (CANDU) - American Machine and Foundry Company (AMF)

Manufacturing - Assembly of components and interconnection of subsystems continued. The fuel and ram tube pressure shell and the handling tube hydraulic manifolds were installed on the cradle. Assembly of the three handling tube shut-off-valves and the shield-plug-and-guide tube internal components continued. Optical alignment of the handling tubes, end fittings and test facility ram was completed. Hookup of the hydraulic power supply to the refueling head was completed.

Difficulties continued in obtaining a Belleville spring with the desired force specifications for the end seal tube. Several sets of springs were fabricated but failed to meet the required load-carrying capabilities when tested. Dimensional and heat treating changes will be incorporated into the fabrication in an effort to obtain a satisfactory set of springs. All manufacturing of components for the on-power refueling head has been completed with the exception of these end-seal springs.

A design change was made in cradle indexing procedures and a new Y-drive gear motor was installed. The motor brake on the unit permits accurate cradle positioning and eliminates the need for positioning to mechanical stops. Resistors were installed in the Y-drive circuit to reduce torque of the new motor at extremes of travel and prevent system damage from occurring if cradle limit switches fail. A single limit switch will control the indexing of the shield-plug-and-guide tube and fine alignment will be accomplished with the cradle being driven in a downward direction. Cradle Z-drive checkout was completed, and final Z-drive limit switch testing was initiated.

Functional checkout of electrical control sequences continued. Changes to the control system, indicated by program checkout, were completed. The Y-index control was changed from 550 to 200 volts to accommodate the new cradle Y-drive. Fabrication of the handling tube-to-control console control switch wiring harness was completed. Bench wiring of the hydraulic manifold-to-transition box harness was started.

Test Facility - Work continued on the water hydraulic control panel and the interconnecting piping to the refueling head; the size of pipe was increased to minimize pressure drops. Thermodynamic testing of the hot loop will be accomplished upon completion of this work. Over-all emergency alert and system shutdown planning continued. Installation of the autoclave safety systems was started. The autoclave safety system will include a bell which will sound at 1600 psi, a relief valve set at 1700 psi, and a safety head containing a rupture disc which will burst at 2600 psi. Work was also initiated on the autoclave water replenishing system.

CANDU Prototype Boiler Section Test

Tests - Combustion Engineering - The data transmitted on the scheduled series of tests on the CANDU boiler were reviewed with the result that more information was desired on several points to better describe the boiler performance.

1. Discrepancies in the heat balance error were noted which left the source of the error unlocalized.

2. The desired level of steam output of 48,000 lb/hr could not be reached in the increased output run due to safety valve limitations.

3. The efficiency of the vortex steam separator in the steam drum appeared to be low.

Combustion Engineering has prepared a cost estimate and proposal for additional tests to obtain further data on the above points.

Development of Sintered Aluminum Products (SAP)

Creep-Rupture Tests - Battelle Memorial Institute - Nine of the 11 long-term tests previously reported in progress were continued through this period. Two of the long-term tests failed with rupture times of 11,341 and 10,685 hr. The tests in progress continued to creep at the approximate rates of previous reports. In most cases these creep rates are changed, generally decreased, only slightly from the rates reported earlier. One of the two failures occurred in Specimen 5-8 from the SAP-895 alloy in test at 10,700 psi and 700°F. This test was stressed for 100,000-hr rupture but failed in 11,341 hr. The specimen overheated about 100°F due to equipment malfunction. Since the other long-term test is still in progress, this test was not restarted. The second failure was on Specimen 75-3 from Alloy M-257 in test at 9,850 psi and 700°F. This test failed on 10,684 hr at very nearly the scheduled 10,000-hr rupture time.

Specimen 6-14 of the SAP-895 alloy in test at 900°F and 7700 psi again showed negative creep during this period. The maximum deformation attained was 0.647% over a time period from 2700 to about 5000 hr. The total deformation now at 15,400 hr is about 0.605% representing a decrease from the previous period when the strain was 0.610%.

Initial results of the creep-rupture tests on the SAP-915 alloy were obtained. As with the other alloys, stresses are chosen to cause failure in the range of 10 to 1000 hr for all three test temperatures--700, 800, and 900°F. Once the stresses are established for these rupture times, their values will be used to choose stresses for failure in 10,000 and 100,000 hr. Some of the tests which were intended for 100 or 1000 hr failure may continue for much longer times.

Non-Destructive Tests

Ultrasonic Testing of Fuel Sheath Tubing - GE, Richland - Fabrication of the prototype 17 mil wall tubing tester has been completed. The entire test station including electronics, mechanical tube scanning system, and analog recorder readout has been assembled. Portions of both the mechanical and electronics system have required redesigning. However, the basic system appears quite workable. The tubing is scanned at 18"/min. One, two, and three mil outside and inside notches were reliably located with fairly linear amplitude relationships.

Header and Feeder Piping Fluid Dynamics

Heat Transfer and Fluid Flow Studies - GE, Richland - Preparation of the following reports is in progress:

1. Comparison of Boiling Burnout in 19-Rod Bundles With and Without Wire Wraps

2. Single Phase and Two-Phase Pressure Losses of Heated 19-Rod Bundles

3. Interim Report on Boiling Burnout With Fog Cooled 19-Rod Bundles.

Plans were made for the two experimental programs now pending. The 6.3' long 0.050" spaced test section which has been used in tests in the horizontal position will be used in tests in the vertical position during February 1964.

Further plans were also made concerning the experiments to be run June 1964 to investigate boiling burnout under fog cooled conditions with a 6' long test section. This test section will approximate as closely as feasible the fuel element and coolant piping chosen by the AECL for a possible fog cooled nuclear power reactor.

Fog Cooled Star Test - Du Pont - The machining of the surrounding rod mockup was completed at the Savannah River Laboratory (SRL). Additional time was taken to hand-fit the auxiliary AC heaters and install the instrumentation before the mockup was shipped September 30 for nickel plating.

Shakedown testing continued with the central rod in a stainless steel housing forming an 0.050" wide annular channel. A 10" spacing of sapphire pins on the central rod was necessary to prevent the central rod from bowing to contact the stainless steel housing. The central rod showed no deterioration after operating one hour at 360,000 Btu hr/ft^2 and 3.6% exit steam quality at 200 psi.

The first runs will be to check the burnout detector operation under subcooled conditions and to determine the indication of the thermocouples at dryout.

Zirconium Technology

Delayed Failure Hydrogen Embrittlement of Zirconium - IIT Research Institute - Specimens of Zr-2.5Nb-0.5 Cu have been produced which exhibit a hydride network (500 ppm hydrogen) in a primary alpha plus transformed beta matrix. This microstructure was obtained by quenching the as-received alloy from 825°C followed by simultaneous aging and hydriding at 353°C. A notched specimen of this material was statically loaded to a stress of 75% of the fracture stress, and unnotched specimens were loaded slightly above and slightly below the yield stress. The notched specimen fractured after 458 hr; however, the unnotched specimens have not failed. The total time at stress has not yet reached the fracture time of unnotched specimens which were hydrided and then heat treated (all transformed beta plus hydrides); a definite conclusion as to the comparative delayed failure susceptibility of this material in the two different microstructural conditions is not yet possible.

An attempt was made to fully heat treat the alloy and then perform hydrogenation at 300°C. This condition would more closely simulate the actual hydrogen absorption in reactor operation, and it was hoped that hydride platelets would concentrate in alpha rendering the material insensitive to static fatigue fracture. After a week in a low pressure atmosphere having a quantity of hydrogen to give 500 ppm in the specimen, most of the hydrogen was absorbed. However, hydrides were heavily concentrated at the surface. The effects of brittle surface layers on the mechanical properties of materials is a topic of considerable importance. Generally speaking, crack nucleation is considerably enhanced. Thus, four specimens of the ternary alloy are being prepared in the manner described above, and delayed failure evaluations will be performed.

Orientation of Hydride Platelets - Du Pont - Work continued on the preparation of specimens for texture measurements by X-ray diffraction. Since the final texture results would not be completed until next month, the primary textures of the tubing were estimated by analysis of previously published texture data and the anisotropy of strain behavior during tensile tests. Present indications are that the susceptibility to stress orientation of the hydride platelets varied widely with little correlation between similar textures and similar fabrication histories.

Creep Tests - Battelle Memorial Institute - The two annealed and two cold-worked specimens were continued in test during this period. The only change in results on these tests from the previous report is accumulated test time, total strain, and perhaps the creep rate. The creep rate, generally, has reached the second stage of constant rate and changes are only occasionally noted. Total test times for the cold-worked material are 49,100 and 51,400 hr for the two tests. Test times for the annealed material are 17,900 for the 550°F test and 25,350 hr for the 650°F test. Two tests were discontinued this period at 650°F and two at 750°F on the annealed material.

Application of the Electron Microprobe to the Study of Zircaloy Corrosion Films - Du Pont - X-ray concentration line scans were performed within the alloy of specimens J-9854 (alpha annealed), J-7871 (alpha + beta annealed), and J-1195 (beta quenched). The results further substantiate those previously reported. The alpha and the alpha + beta annealed specimens contain two types of precipitates containing Fe-Ni-Zr and Fe-Cr-Zr, respectively. The Sn concentration within these precipitates has not been determined as yet.

X-ray area scans of specimen J-1195 (beta quenched) previously reported the absence of nickel in some of the iron-chromium rich bands. X-ray concentration line scans, which are more sensitive to concentration variations, show that increases in nickel concentration occur coincidently with those of iron and chromium, but at a much lower level.

Heat Transfer at Low Heat Flux

Tests - University of Michigan - There were no satisfactory test runs made during the month. Difficulty of obtaining reproducible data and the presence of black magnetic iron oxide (magnetite) powder in the low heat flux boiling system prompted a close look at the dissolved oxygen content in the water. As previously reported, a layer of magnetite has been found on the test electrode. With this a higher tube surface temperature is required to effect a given heat flux than is required for a clean surface. Investigation of the literature reveals that for high pressure systems the dissolved oxygen content must be held much below the level required by low pressure systems and that boiling to degas is probably inadequate to obtain the required level. It is considered that by monitoring the oxygen level and using an oxygen scavenger such as hydrazine, the problem of loose magnetite powder in the system can be controlled.

The laboratory constant temperature block assembly was found to be in need of repair prior to further use in calibrating thermocouples for test electrodes; repair has been initiated. This assembly is also used in making electrical resistivity measurements on the 3/4" tube samples at high temperatures.

The installation of a series of switches to allow use of a four channel Sanborn recorder for simultaneous recording to the thermocouple voltage was completed. In preliminary operation checks the recording system appears to perform satisfactorily. With proper technique in recording, its accuracy will be within approximately ± 0.1°F.

Irradiation Programs

CANDU Fuel in Heavy Water Components Test Reactor (HWCTR) - During October, the HWCTR operated at an average power of 46 MWt and a system pressure of 1200 psig.

I. RESEARCH AND DEVELOPMENT (CONT.)

Maximum exposure accumulated to October 31 was 4310 MWD/T.

Time at temperature totals 4640 hr.

Organic Loop - Massachusetts Institute of Technology - Irradiation of Santowax WR commenced July 24, 1963, and has been proceeding well since that time. Although the start of irradiation was delayed approximately three weeks due to late delivery of the Santowax WR from Monsanto, MIT is close to the desired schedule as outlined to meet our objectives.

As of September 15, the irradiated coolant contained approximately 40 w/o degradation products. Physical and chemical analysis of the coolant is proceeding as the irradiation progresses.

D_2O Technology

Sieve Tray Capacity Tests - Du Pont - Equipment in Units 31 and 32 of Building 413 will be modified and operated to test the capacity of sieve trays under GS process conditions. Since the total unit is tied directly to H_2S makeup, entrainment measurements will be made at an operating pressure of 260 psig currently standard in Building 412.

Shop work and field modifications are in progress. Construction is 20% complete. Scheduled completion is December 16, 1963. Hydrostatic testing, flushing the equipment, and a dummy run with nitrogen gas will be made prior to initiation of tests.

Process Development Pile, Du Pont, SRL - A program has been approved to make measurements in the PDP of a core simulating a partially burned up reactor core. This would be done by simulating the concentration of U-235, U-238, Pu-239, Pu-240, etc., characteristic of a fuel with an exposure of a specific number of MWD/T. It has been determined that 8% Pu-240 in Pu-239 is the highest available. Work on preparation of specifications and other procurement action is being undertaken, so that when the 8% Pu-240 becomes available fabrication of the fuel for this experiment can begin immediately.

DU PONT

Large Reactor Plant Studies - Information requested by the Interagency Committee on Desalination was compiled on R&D and construction costs for three different size reactor plants. These were: a single reactor power plant of 750 MWe to be completed by 1968; a 1500 MWe single reactor power plant to be completed by 1974; and a 8300 MWt (2250 MWe) power plant to be completed by 1980.

Physics Experiments for the French EL-4 - SRL - Reactor physics experiments are underway in the Process Development Pile in support of the French EL-4 reactor. These experiments will determine radial and vertical buckling with both D_2O and air in the housing tube. The magnitude of the vertical migration area and the lattice anisotropy (the ratio of vertical to radial migration area) will be determined. Information will also be obtained on reflector effectiveness in lattices containing a large void volume. A French observer is present.

Thorium Breeder Reactor Program - A preliminary outline of a proposed development schedule has been prepared. The most significant item at present is for irradiation of a thorium fuel element in the Heavy Water Components Test Reactor (HWCTR). It is estimated that it will take nine months to design and procure the first element for irradiation from the time authority is given to proceed.

I. RESEARCH AND DEVELOPMENT (CONT.)

Oxide Elements - Development work was started at SRL on the oxide elements SOT-6 and SOT-9 for irradiation in HWCTR. Swaging dies had to be reworked to accommodate the new diameter (2.5" OD) tubes because of springback. The SOT-6 is expected to operate at $\int k d\theta$ 30 and SOT-9 at $\int k d\theta$ 40. The SOT-6 assembly will contain natural uranium and SOT-9 uranium enriched to 1.2%.

COLUMBIA UNIVERSITY

Three test sections are planned for the next run in the Task XIII loop to study the hydraulic interaction between parallel heated channels. Size of the sections are as follows:

Single tube	-	0.5" ID	76" long
Single tube	-	0.25" ID	40" long
Three parallel tubes	-	0.5" ID	76" long

The single 0.5" tube will provide a reference point at 1500 psi for the parallel channels and to provide a basis for comparing previous data obtained at 1000 psi. The 0.5" ID test section was prepared and scheduled for testing the first week in November. The 0.5" ID three parallel tube test section was also being fabricated.

A bypass line was installed in the loop to permit control of the degree of interaction between the parallel channels. There are two limiting conditions with respect to coolant flow to the assembly that can exist; these are constant ΔP and constant volumetric flow. Most operation on Task XIII has been close to the constant volumetric flow type. The addition of the bypass will permit variation in the direction of constant ΔP.

The single 0.25" ID tube which was being designed will be used to extend existing data to smaller diameters.

NUCLEAR METALS INC. (NMI)

NMI has been engaged in a program designed to produce a new driver fuel for the HWCTR. The driver fuel designs consisted of two concentric tubes spaced with ribs. After proving the concept and making a preliminary cost proposal, it was decided that the benefits to be derived from a two tube driver design was not warranted because of the high cost of producing these tubes. The development program at NMI was redirected to a single tube driver fuel concept similar to the driver fuel now in HWCTR but with a higher enriched uranium content.

II. HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

Operations Summary - October 1963

Reactor Innage	81.5%
No. of Scrams and Shutdowns	2
Maximum Power	53 MWt
Moderator Temperature	200°C
System Pressure	1200 psi
Monthly Accumulated Exposure	1107 MWD

II. HEAVY WATER COMPONENTS TEST REACTOR (HWCTR) (CONT.)

Monthly Losses:

D_2O (100 mol %)	622 lb
Helium	68185 scf
Deuterium	4116 gm

Operating History - Operation of the HWCTR was resumed October 3, following replacement of the oxide fuel assembly which failed September 28. The reactor was shut down October 4 due to excessive oxygen concentration in the Liquid Loop. The oxygen was brought within limits by venting the gas system through the Liquid Loop seal head tank vent line.

Nuclear operation was resumed October 4, and continued until October 29 at power levels up to 53 MWt, the highest power level achieved to date. A manual scram was initiated following a rapid increase in activity as detected by the Liquid Loop activity monitors. An increase from gaseous fission product activity was subsequently detected by the building air monitors. The reactor was cooled down and depressurized. The Liquid Loop system has been purged intermittently to reduce the activity in the system.

Discharge of the failed fuel assembly in the Liquid Loop was delayed until November 3 to permit the building activity level to decline. The discharge was accomplished on that date and the assembly was stored in a failed fuel container in the spent fuel basin. The failed assembly consists of seven tubular fuel segments of natural and 0.91% enriched UO_2 which had accumulated a total exposure of 1075 MWD/T at a maximum $\int k d\theta$ of 63 watts/cm.

During the operating period prior to shutdown, the segmented tubular test assembly of 1.5% enriched UO_2 which was charged to the reactor with the initial test change (October 1962) exceeded a maximum cumulative exposure of 10,000 MWD/T. This assembly has been operating at an $\int k d\theta$ of 22-28 watts/cm.

It is anticipated that nuclear operation will be resumed about November 5 or 6, following replacement of the deionizers in the Liquid Loop system.

CANDU Assembly Data

The following CANDU assembly data is based on HWCTR power of 51 MWt, moderator temperature of 200°C, and system pressure of 1200 psig:

Assembly Power	1.38 MWt
Assembly Coolant Flow	137 gpm
Peak Heat Flux	180,000 pcu/hr ft^2
Max. Specific Power	32 MW/T
$\int k d\theta$	39 w/cm
Max. Total Accumulated Exposure	4310 MWD/T
Reactor Scrams and Shutdowns	2
Time at Temperature during October	580 hr

There were no charges or discharges of the CANDU assembly during October.

Current Test Fuel Loading (As of October 31)

Position	Fuel Type	Acc. Exposure MWD/T (max.)
37	CANDU, Nat. UO_2 19-rod bundles	4,310
38 (Liquid Loop)	Seg. tubes of nat. and 0.91% enr. UO_2 (Discharged 11/3)	1,075
39	Single tube of unalloyed 2.1% enr. U-metal	5,980
40	Segmented tubes of 1.5% enr. UO_2	10,950

II. HEAVY WATER COMPONENTS TEST REACTOR (HWCTR) (CONT.)

Position	Fuel Type	Acc. Exposure MWD/T (max.)
41 (Boiling Loop)	Empty	--
42	Single tube of unalloyed 2.1% enr. U-metal	5,750
55	Single tube of 1.5% enr. UO_2	8,580
56	Segmented tubes of 1.5% enr. UO_2	8,800
57	Segmented tubes of nat. U-Fe-Al alloy	2,160
58	Segemented tubes of 1.5% enr. UO_2	4,410
59	Single tube of 1.5% enr. UO_2	4,130
60	Segmented tubes of nat. U-Fe-Al-Si Alloy	2,100

III. POWER DEMONSTRATION PROGRAM

CAROLINAS-VIRGINIA TUBE REACTOR (CVTR)

The major effort during October was directed to restoring the header cavity and rod drive trenches to an operable condition. All of the Mirror insulation was installed on the header cavity. Because of the flourides contained in the flux solution used in resoldering joints in the insulation, tedious washing and testing for flouride solution was required. Upon completion of the installation of the Mirror insulation the components removed from the header cavity were reinstalled. These included rod drive shafts, pinion bodies, flow impulse lines, thermohms and the flux wire system. The rod drive cables, potted with Scotchcast epoxy were also installed and satisfactorily leak tested. An over-all leak test of the header cavity system and the rod drive trench system was conducted. At 2" and at 10" water pressure on the header cavity leakage was reported at 0.5" and 0.7"/hr, respectively. This correlates to approximately 300 to 400 scf helium loss/day, which is acceptable. Reinstallation of the rod drive units required reshimming and realignment. Tests on the flux wire system after installation showed the system to be operable and acceptable.

The shutdown coolers and the component cooling water heat exchangers, which were returned to the vendor for modification and repair, have been reinstalled.

The refueling machine was partially tested out. A U-tube was removed from the core and new Conoseals were installed on the U-tube and then the U-tube reinstalled in the core. The machine proved to be satisfactory except for positioning of the refueling machine. With the machine positioned on previously mapped coordinates the U-tube could not be picked up with the machine. Deviation appeared to be approximately 1/2" in each direction. This problem is being investigated.

The initial installation of orifices for the CVTR core was made.

Two criticality runs with low moderator height were made in order to build up photoneutrons for detection by startup instrumentation.

The primary loop was filled and the primary pumps jogged and vented. The U-tube flow impulse lines were filled and a cold test was made at 110% of operating pressure. The test showed no leaks in the primary system.

A sample of D_2O was taken and analyzed from the primary system; purity was 99.71%. This was the first sample secured over a period of months and shows no degradation during the shutdown period.

Initiation of the Research and Development thermal and hydraulic test was made after all instrumentation had been checked out and calibrated. One objective of this test is to determine the effect on individual U-tube flows by installing orifice plates in specific U-tubes, then varying the size of these plates in two of the U-tubes. The second objective is to determine the conducted heat loss to the moderator from the primary loop under various conditions of operation with no nuclear heating of the moderator. Heat was applied to the primary system either from an auxiliary boiler transferring heat to primary system from the main heat exchanger or from steam supplied by the Parr Steam Plant. The first part of the test relating to initial orifice placement was conducted, heating the moderator to 465°F from ambient temperature and taking measurements every 50°F. The data have not been analyzed but no abnormal conditions were observed. On the cooldown of the primary system there was a spurious signal from gas monitor RIC-9, caused by a short circuit in two of the plug contacts. This caused the reactor to scram and moderator to dump. The condition was corrected, the moderator returned to the moderator tank and the second portion of the orifice tests is now in progress.

Preliminary analysis of previously conducted cold rod worths gives the following results:

1. The differential moderator worths can be related to the height of the moderator above the bottom of the fuel by the equation

$$\left(\frac{\Delta k/k}{\Delta H} \times 10^4\right)^{-1/3} = .00500H + .06933$$

2. Integral control rod worths calculated by integration of the differential moderator worth curve between critical moderator heights at ~90°F given in units of $\Delta k/k$ are:

Total Control Rod Worth	$\Delta k/k$ = 27.4%
Installed Reactivity	$\Delta k/k$ = 21.0%
Shutdown	$\Delta k/k$ = 6.4%

3. The worth of any group of control rods is very dependent on the sequence in which it is withdrawn. In general, however, the control rod worths are close to those calculated for the CVTR design.

Work continued on debugging and modifying the CVTR flux wire data reduction code and on generating the required nuclear parameters to translate flux wire activity into assembly power. A number of modifications were incorporated into the code to increase the versatility of the program and make it capable of accepting a greater variety of flux wire run combinations.

The PDQ studies to determine the nuclear factors required to translate flux wire activity into assembly power were continued.