APDA-123

QUARTERLY REPORT

OF

RESEARCH AND DEVELOPMENT ACTIVITIES

April-June 1957

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INTRODUCTION

This is the first of a series of quarterly progress reports to be prepared covering the research and development activities performed or sponsored by APDA in support of the Enrico Fermi Atomic Power Plant. The design and engineering work for that plant is not covered in this report.

It must be recognized that work has been carried on since 1951 on the liquid-metal-cooled fast breeder reactor; consequently the background information herein is sketchy. However, in preparing this report, an attempt was made to present sufficient background on the items reported so that these items would be meaningful. It is intended that background information will be given with each new item; however, such information generally will not be repeated in subsequent reports on the same item.

It is expected that a complete description of major research and development programs and the detailed results therefrom will be contained in APDA topical reports or in articles prepared by APDA staff members for publication in the scientific and technical press. Reference will be made to such reports whenever possible in order to avoid repeating the information in these quarterly reports.

NUCLEAR ENGINEERING

Most of the APDA effort in nuclear engineering during this period involved no research and development; effort was primarily devoted to the calculations and the checking of designs for facilities to be incorporated into the Enrico Fermi Atomic Power Plant. The work is reported because of its possible interest and because it presents background information.

SHIELDING

The shield system for the Enrico Fermi reactor is composed of a primary neutron shield consisting of steel, graphite, borated graphite, and carbon block; a secondary shield of steel and concrete to prevent secondary coolant activation; and a biological shield of concrete to reduce the final dose level to 30 mr per 40-hour week in areas of the reactor building that will be inhabited during operation.

A rotating plug forms the top closure of the reactor and supports the control mechanisms, fuel handling mechanisms, and fuel subassembly hold-down apparatus. This plug is designed to consist of alternating layers of steel, borated steel, and borated graphite. In an attempt to reduce the price of this plug, estimates were prepared on an alternate design in which a mixture of ferroboron and carbon was used to replace the boron steel and borated graphite. This estimate showed that the ferroboron-carbon mixture was about one-third as costly as the solid heterogeneous scheme. However, the unknown characteristics of the ferroboron-carbon mixture, particularly in respect to segregation due to vibration and shrinkage, led to the decision to retain the present design. Consideration is being given to a research program directed toward answering these problems, since the ferroboron seems to have the potential of allowing considerably reduced shielding costs.

A report, KLX-1782, "Inert Gas System Hazards Analysis," was received from Vitro Corporation of America as a result of work performed by them under contract to APDA. That report summarizes the problem of activation of the inert cover gas over the sodium system and proposes gas system designs capable of effectively handling the problem.

The design of the secondary shield received considerable attention. The total heat generation in this shield due to gamma absorption is about 100 kw. The design consists of 30 inches of reinforced concrete faced on the inner (reactor) side with steel thermal shields which vary in thickness from 0.5 to 2.5 inches, depending on the proximity and intensity of the radiation sources. These thermal shields are designed to reduce the gamma energy flux at any concrete surface to 2.1×10^{10} mev/cm^2/sec and the maximum temperature of the concrete to 200 F.

It has been found that if the lower reactor compartment atmosphere is cooled to 90 F, natural circulation of this atmosphere up through the gap between the concrete and the steel thermal shield results in adequate cooling to maintain the 200 F limitation.

Battelle Memorial Institute has made a study of the primary shield for the Enrico Fermi Reactor in which temperature distributions, irradiation effects, stacking arrangement, voidage, and economics for the borated graphite shield were investigated. A report describing this work was issued as BMI-APDA-622.

In conjunction with the secondary shield cooling study, an estimate was prepared of the expected total heat generation within the lower reactor compartment. This total was found to be about 1.7 x 10^6 Btu/hr, of which 60% was due to absorption of neutron and gamma radiation and 40% was due to heat loss into the region from high temperature piping and other components.

REACTOR PHYSICS

In conjunction with the development of economical fabrication procedures for the fuel subassemblies, estimates were prepared of the amount of core material which can be safely handled as a unit from the standpoint of criticality. It was found that for the 25% enriched uranium 10 w/o molybdenum fuel material, the spherical critical mass for a well-reflected assembly is 200 kg of total material. This figure was obtained from diffusion theory and transport corrections and agrees with the fragmentary experimental evidence. In the absence of moderating materials, this amount appears to allow considerable freedom for the fabrication process.

Estimates have been made of the effect on plutonium production of alloying the breeder blanket. It has been found that if molybdenum, the core alloying material, is also used in the blanket, a reduction in conversion ratio occurs equal to 0.012 for each weight per cent of Mo in the blanket. The economic loss from this reduction in Pu production appears to be more than offset by increased blanket lifetime, concomitant with the increased radiation stability due to the Mo addition.

Since some consideration has been given to the use of tantalum as a fuel cladding material, estimates of the effect of Ta on critical mass and conversion ratio were carried out using the scanty cross-section data available and filling the omissions with theoretical predictions. It was found that replacement of the zirconium clad with tantalum reduces the conversion ratio by nearly 10% and increases the critical mass a like amount.

A large number of multigroup-multiregion physics calculations were completed both in connection with the above-mentioned studies and with routine investigations into other aspects of the design.

REACTOR CONTROL AND KINETICS

APDA is furnishing the technical direction for a program at the Holley Carburetor Company covering the analyses of the transient characteristics of the reactor, primary coolant system, intermediate coolant system, and steam system. This work is being performed under contract to PRDC. Holley completed most of the early phases of the study which consisted of preparation of a detailed mathematical model of the plant and detailed analysis of the steam components. Preparation of analog and digital computers for the actual overall analysis was begun.

APDA prepared a detailed outline of the reactor power limiting system; and, based on this outline, Bendix Aviation Corporation has started work on the simulation studies and design engineering required to establish control and instrumentation specifications for the required equipment.

APDA completed preparation of a simulator representation of the EBR-I blanket heat transfer situation. A number of simulator runs were made in the continuing effort to provide a rational explanation of the dynamic oscillations observed in EBR-I.

Under contract to APDA, Nuclear Development Corporation of America continued investigation of the course of supercriticality accidents caused by reactor core melting in the hypothesized absence of sodium.

MATERIALS

FISSIONABLE MATERIAL STUDIES

Radiation Stability - This program covers the dimensional stability of reactor materials when exposed in nuclear radiational fields. Uranium binary alloys, ternary alloys, and certain matrix elements or compounds are being investigated.

The reference fuel alloy, U-10 w/o Mo, has been irradiated to burn-ups comparable to that expected during plant operation. Preliminary tests have already been made beyond 1 a/o burn-up. Proof tests are under way to reach burnups of 2% and 3%. The radiation stability program has included tests on fuels irradiated at controlled temperatures.

The APDA irradiation program from 1954 to June, 1957, is described and evaluated in report APDA-122, "The APDA Irradiation Test Program on Selected Fuel Alloys."

Corrosion Tests - This work includes operation of forced circulation sodium loops under isothermal conditions for both short and long-time corrosion tests on full-length fuel pins and a series of fuel pin samples. At Babcock & Wilcox, 21 zirconium-clad fuel pin specimens with welded, swaged, and pointed end-closures exhibited no corrosive attack after a 1000-hour exposure to flowing sodium at 1000 F in a corrosion test loop. A tantalum-clad fuel pin without end-closures behaved similarly. A metallurgical examination was begun on each specimen to determine the extent of sodium penetration, if any, at the pin end-closure area.

Heat Treatment Effects - At General Motors, a program to evaluate various procedures for production heat treatment for the reference fuel element was essentially completed. This program covered the investigation of heat treatments in vacuum and inert gas atmospheres with the evaluation being based on microstructural and physical examinations. Results of this program were as follows: (1) a satisfactory metallographic structure is obtained on zirconium-clad U-10% Mo alloy heat treated for 1 hour at 800 C in either vacuum or argon atmosphere and slow cooled from 800 C to 200 C, and (2) the use of vacuum during this heat treatment rather than purified argon results in less embrittlement and oxygen contamination of the zirconium clad. Effects of post-fabrication heat treatment on radiation stability of the U-10% Mo alloy is currently being tested.

Evaluation of Fuel Fabrication Processes - Reference fuel pins in various stages of fabrication — as-extruded, partially-swaged, and fully-swaged — were examined metallographically at General Motors. Pins produced by both Babcock & Wilcox and Nuclear Metals were included in this study. Observations made during this work were as follows: (1) in all stages of fabrication, pins produced by B&W appeared to be more nearly homogeneous than pins produced by NMI, (2) the whorl pattern observed on fully-swaged B&W pins was not evident on as-extruded or partially-swaged B&W pins nor was it evident on any of the NMI pins. Fully-swaged B&W pins represent a 73% reduction of area, whereas fully-swaged NMI pins represent a 48% reduction of area. The effect of fabrication variables on radiation stability is currently being evaluated by radiation tests.

Metallurgical examinations of fuel pins have been carried out by Detroit Edison. A report on this subject, EL & RD 57B52-1, entitled, "Metallurgical Examination of Three Uranium-10 Weight Per Cent Molybdenum Fuel Rods to Determine Conformance to APDA Specifications," March, 1957, has been published.

Thermal Cycling Tests - At Battelle, 2 cermet specimens of UO_2-10% Mo-U were thermal cycled in NaK between 100-700 C. After 10 cycles, run in a 14-day period, no changes were noted during preliminary examination. Although the specimens were not visably affected, analyses of the NaK to determine both soluble and insoluble particles, as well as MoO_2 formation, are in progress.

STRUCTURAL MATERIAL STUDIES

Corrosion Tests - Previous test work was directed mainly toward the Type 18-8 austenitic steels. These initial tests indicated a need for test rigs closely duplicating actual operating conditions, and the economics of structural materials indicated the need to investigate steels other than the Type 18-8 stainless steels. In a test at General Motors that was designed to evaluate the effect of a water leak into the sodium system, specimens of stainless steel and 2-1/4 Cr-1 Mo steel were exposed to molten sodium containing up to 6% NaOH. The test results showed: (1) uniform corrosive attack on the Type 403 stainless steel, and (2) no attack on the Type 304 stainless steel or the 2-1/4 Cr-1 Mo ferritic steel. The specimens were exposed at temperatures ranging between 750 F and 950 F in thermally circulated sodium. It is anticipated that an additional 6-week test will be started with 6% NaOH concentration and with temperatures ranging between 550 F and 850 F to confirm results previously obtained.

Self-Weld Studies - A number of ferrous and nonferrous materials have been tested for galling and self-welding tendencies while in mutual contact under pressure in liquid sodium. The program is a continuation at Allis-Chalmers of test work on selected materials with the desired properties.

A plain carbon-steel, which is used as a standard for rating other steels for machinability, did not gall against Type 347 stainless steel under unit pressure loadings of 1000, 2500, 5000, and 10,000 psi in sodium at 500 F to 1000 F. The surfaces of both metals were polished as a result of the test and were only slightly deformed. Hard-surfaced specimens of nitrided Type 347 stainless steel, post-heat-treated to relieve internal stresses and provide a more ductile surface, were tested in sodium with pressure and temperature loadings to 10,000 psi and 1000 F. The surface-treated specimens underwent no wear or self-welding when hand rotated against themselves and against bare Type 304 stainless steel. These tests substantiated the suitability of nitriding as a means to prevent wear, as indicated by several previous tests.

Effects of Irradiation - APDA is cooperating with Argonne National Laboratory in the examination of the stainless steel hexagonal can that surrounded the core of EBR-I during its lifetime. The purpose is to obtain additional information on the effect of irradiation of stainless steel, particularly from fast neutron exposure. Projected tests, which will be done by remote control, incl 's room temperature and hot hardness tests, tensile tests, and tests to determine the effects of isothermal anneals on tensile strength and hardness.

Dry Lubricant Coatings - In an effort to avoid the galling and self-welding tendencies of bearing metals in direct contact with each other in dry gas or sodium-vapor atmospheres, self-lubricated surface coatings are being studied. During preliminary tests at Allis-Chalmers, two Type 304 stainless steel specimens that were coated with boron nitride in a nickel matrix suffered some galling when tested against each other at 500-800 F in an argon atmosphere with a unit pressure loading of 1000 psi. One specimen had a 10% BN-90% Ni coating and the other specimen had a 20% BN-80% Ni coating.

Compressibility of Graphite and Carbon - Restraint-compression tests on carbon and graphite were performed by Detroit Edison at a maximum pressure of 75,000 psi to determine the shock absorbing qualities of these materials. Both materials behaved elastically, with graphite being more elastic than carbon or pine wood that was used merely for comparison. Since increasing load rates from 0.025 inch/min to 0.25 inch/min decreases deflection by 20%, dynamic tests on the same materials are planned. Tests are also being expanded to cover low density materials.

PUBLICATIONS

A paper entitled, "Designing and Fabricating the APDA Nuclear Reactor Vessels," was presented before the American Welding Society in Philadelphia on April 9. This paper, prepared jointly by APDA and Combustion Engineering, describes the design considerations, welding procedures, and material selection for the Enrico Fermi reactor vessel.

A paper entitled, "Effect of Heat Treatment and Burn-up on the Radiation Stability of 10% Mo-U Fuel Alloys," was presented jointly by APDA and Battelle at the annual meeting of the American Society for Testing Materials in Atlantic City on June 20. This paper contains a summary and evaluation of the results of the APDA-sponsored irradiation program at Battelle.

FUEL ENGINEERING

FUEL PINS

Fabrication - The fuel pin to be used in the Enrico Fermi Reactor is a coextruded, metallurgically-bonded, zirconium-clad pin with a 0.004-inch clad. The pin is cold swaged to its final diameter of 0.158 inch, is 30.5 inches long before end-capping, and contains a U-10 w/o Mo alloy. Research and development work is being done at both Nuclear Metals and Baboock & Wilcox on the fabrication of these fuel pins.

End-Capping - Development of end-caps for the fuel pin is being done at Battelle Memorial Institute and Nuclear Metals. At Battelle, an inert-gas-shielded, tungsten-arc weld was used to seal a mechanically-locked end-cap. Owing to the thin wall of the cladding and the fuel pin manufacturing tolerances, high quality welds were not attainable; therefore, this effort was discontinued. Attempts were also made to seal the extreme tip of fuel pins that had been swaged to a point. Inert-gas-shielded, tungsten-arc welding was tried on specimens that had zirconium wire inserted in a drilled hole at the point. However, the core metal alloyed with the zirconium, causing uranium to appear at the surface; therefore, this method also was discontinued.

Attempts to seal the extreme tip of pointed fuel pins by dipping them in molten zirconium proved to be unsuccessful. Metallographic examination indicated that alloying takes place, allowing uranium to appear at the surface. Also, the bond at the seal is not satisfactory.

At Nuclear Metals, additional specimens of zirconium-clad fuel pins were swaged to a point in an effort to seal them by use of swaging dies that were elliptical or parabolic in shape. Preliminary results were no more promising than earlier swaging efforts using cone-shaped swaging dies. Further specimens will be pointed with the existing dies, and a more complete analysis of the process will be made.

Since it is not economically feasible to fabricate a fuel pin that is hermetically sealed, it was necessary to calculate the maximum gap that could be tolerated for an end-cap which is mechanically locked to the ends of the fuel pins. This gap is the annular space between the outside diameter of the pin and the inside diameter of the end-cap after the end-cap has been swaged onto the fuel pin. The basis for this calculation was: (1) coolant entering the reactor at 550 F and leaving the core at 1015 F, and (2) a design limitation of 1300 F was used for the central metal temperature of the hottest fuel pin. It was found that the core alloy will not reach the hot spot design limitation of 1300 F at the anchor end of the fuel pin if the maximum gap between the pin and a Type 304 stainless steel cap is 0.0005 inch, even if the gap is filled with only low conductivity fission product gases. Furthermore, it was found that the gap can attain a thickness of 0.0007 inch if a zirconium end-cap is used instead of a stainless steel end-cap at this point. It was found that a maximum allowable gap thickness of 0.005 inch and a zirconium end-cap may be necessary at the hot, or outlet, end of the fuel pin in order not to exceed the 1300 F hot spot design limitation.

MELT-DOWN STUDIES

A program at Nuclear Metals is being carried out in which the integrity of the zirconium clad of the fuel pin is observed when the uranium alloy is heated above its melting temperature. Single-pin melt-down tests were concluded with the testing of an additional 11 fuel pins having a clad thickness of 0.010 inch; a total of 46 single-pin tests was conducted. Results showed that, in general, the heavier the clad material, the longer the pin will contain molten fuel alloy. It was also found that tantalum and zirconium spacer wires between pins do not lead to early failure, whereas stainless steel spacer wires do because of the resulting low-temperature Fe-Zr eutectic. The experiments demonstrated that die marks or other surface defects or impurities also cause early failure.

STRESS-STRAIN DATA

Stress-strain tests at elevated temperatures were run at Nuclear Metals on sponge zirconium specimens which had a reduction in area by extrusion and by swaging equivalent to that of the zirconium-clad, reference design fuel pins. The heat-treated specimens were pulled at temperatures of 250 F and 500 F; additional specimens will be pulled at 750 F and 1000 F. The data obtained is being analyzed.

THERMAL CYCLING

Since cyclic temperatures which result from normal start-up and shut-down operational procedures could cause premature fuel pin failures, a thermal cycling test program is being carried out by Detroit Edison. Two possible mechanisms of failure are breakdown of fuel pin cladding by rubbing action of adjacent spacer wires and severe bowing of the fuel pins that will block off coolant flow passages. One thermal cycling test on a single zirconium-clad, reference fuel pin has been completed. The pin, which was supported laterally by spacer wires so that both the effects of thermal cycling on a reference pin and the rubbing effects of the spacer wires on the zirconium cladding could be determined, was subjected to 25 thermal cycles in hot sodium between 225 F and 1400 F. Preliminary examination showed no evidence of dimensional changes, distortion, or surface damage at the point of contact with the spacer wires. Plans were made to thermal cycle a cluster of nine reference fuel pins fabricated by Babcock & Wilcox and Nuclear Metals; heat treatment, surface preparation, and end-capping will vary on the different specimens. Such a test will determine whether any of the variables considered will result in a particular pin exhibiting superior behavior under thermal cycling. The fuel pins for this test are now being fabricated.

Two reports have been issued on thermal cycling tests by Detroit Edison: EL & RD 56E30 "Thermal Cycle Test of an APDA Fuel Pin," dated January, 1957, and EL & RD 57C05-1 "Thermal Cycle Test in Sodium of an APDA Fuel Pin," dated June, 1957.

HYDRAULIC TESTS OF CORE AND BLANKET SUBASSEMBLIES

Before the design of the core and radial blanket subassemblies for the Enrico Fermi Reactor could be frozen, it was necessary that hydraulic test data be made available to determine the head losses, overall pressure drop, velocity distribution, mixing effects, mechanical rigidity under dynamic fluid forces, and orifice sizes. Both analytical and experimental work is being done at the Engineering and Research Institute, University of Michigan, on this program. Pressure drop data from hydraulic tests on a streamlined core subassembly showed that the loss through the inlet nozzle, transition zone, lower blanket, upper blanket, and handling lug, excluding the core section, amounts to 27.7 psi at rated flow. The reduction of the losses as a result of streamlining the internals of the subassembly amounts to 16%. Two reports describing these tests that were published by the university are, "Tests on Models of Nuclear Reactor Elements" - I, Head Losses in Blanket Subassembly, ERI 2431-1-P, March, 1956; and II, "Studies of Diffusion, "ERI 2431-2-P, March, 1957.

IRRADIATION OF FUEL PINS

In addition to radiation test discussed earlier, two separate irradiation programs have been initiated to determine the irradiation stability of reference size fuel pins and to determine the irradiation stability of 49-pin assemblies that have hardware and fuel pin spacing, length, and diameter similar to that in the reference 144-pin subassembly of the Enrico Fermi reactor. The full-length pins will be irradiated in the CP-5 reactor at ANL and the 49-pin assemblies will be irradiated in the SRE at Atomics International.

The first full-length fuel pin containing enriched fuel alloy that was fabricated for irradiation in the CP-5 reactor was sent to Babcock & Wilcox to be encapsulated. The apparatus which was designed to sodium bond the reference APDA blanket rods was used to NaK bond the irradiation specimen to the finned capsule. Babcock & Wilcox also installed a number of thermocouples on the capsule. After encapsulation, the pin was shipped to ANL and inserted into the specially constructed in-pile closed loop where it will remain until 1% total atom burn-up is obtained. A second closed loop for irradiating APDA reference-design fuel pins is under construction and is 75% complete. The fuel pin specimen for this second loop is already fabricated and is scheduled to undergo an irradiation to 1% total atom burn-up.

CORE-II PROGRAM

The APDA Core-II program has progressed under contract with Battelle Memorial Institute since September, 1956. A major objective of this five-year design and development program is to provide an improved fuel system for the Enrico Fermi Reactor. The initial program consists of an evaluation and design study to determine the most promising fuel system to develop in order to meet these objectives.

The selection of fuel materials for Core-II was reduced first to four general types and subsequently to three types by Battelle. The three systems to be subjected to concentrated economic analyses and detailed design studies are: (1) U-10% Mo alloy plates metallurgically bonded to zirconium clad, (2) stainless-steel-clad pins containing UO_2 or UO_2 dispersed in sodium, and (3) zirconium-clad cermet of UO_2 dispersed in U-10% Mo alloy plates. The fourth system that was considered but was subsequently dropped because a breeding ratio of less than unity would result within the Core-II design ground rules was a stainless-steel-clad cermet of UO_2 dispersed in a stainless steel matrix.

Calculations of the Doppler effect in cermet fuels for a fast reactor indicate that the UO_2 portion may be fully enriched in a U-238 alloy matrix for

UO_2 particles 400 microns in size or smaller. Since the minimum U-238 to U-235 ratio that will be permitted in the Enrico Fermi Power Plant reactor for intimately mixed fuel has been fixed because of reactivity coefficients, Battelle was attempting to determine by heat transfer analysis whether this ratio was to be maintained on an atom basis or on a particle basis. The conclusion was that the ratio could be maintained on a particle basis.

During the evaluation and design study to determine the most promising fuel systems, alloys of niobium and zirconium were also considered. It was determined that, for equivalent effects on breeding ratio, zirconium contents in fuel alloys as high as 40% to 50% by weight could be tolerated while molybdenum and niobium were limited to 10% to 15% by weight.

REACTOR VESSEL DESIGN

ROTATING PLUG BEARING TEST

At Franklin Institute, a scale mock-up of the bearing for the rotating plug was loaded and is being rotated under high temperatures simulating the operation of the plug under nonnuclear reactor operating conditions. The purpose of the test is to determine the best lubricant for the bearings. A report on this test is expected by the end of the year.

COMPATIBILITY TESTS

At Allis-Chalmers, compatibility tests are being made on core subassembly nozzle to determine (1) whether the clearances between the nozzle and nozzle sleeve and between the nozzle sleeve and support plates are adequate, and (2) the compatibility of materials used in the nozzle, sleeve and plates. A description of these tests and the results therefrom are contained in Allis-Chalmers Report No. ACNP-5728.

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TEST OPERATIONS

NaK-WATER REACTION TEST

Introduction - Two years ago tests were started to study the reaction of NaK and water under conditions simulating a tube rupture in a steam generator for the initial steam and NaK conditions expected in the Enrico Fermi Plant. The study was divided into two phases:

Phase-I - Fundamental tests of a 0.5-inch-diameter leak of water into NaK with varying amounts of inert gas expansion volume in the NaK system. The NaK system was a "closed" system, having no pressure relief.

Phase-II - Tests of a 0.5-inch-diameter tube rupturing in the center of a 7-tube heat exchanger bundle. This test embodies variations in the "closed" system concept with inert gas volume and the "open" system equipped with a low-pressure relief to atmosphere.

The Phase-I tests were completed prior to April 1957, and a report on these tests is in preparation.

Phase-II Tests - At the conclusion of the first three runs of Phase-II, made prior to April, in which ruptures of a steam generator tube were simulated in a seven-tube heat exchanger, there was no visual evidence of damage to adjacent tubes. Runs four through seven were made with considerable difficulty due to poor NaK flow and to the valve stem freezing in the packing gland of the trigger valve because of the formation of NaK oxide.

The NaK pump was reinstalled in the loops, and Runs 8, 9, and 10 were attempted with forced NaK circulation. This pump is a mechanical pump, and considerable difficulty was experienced with the packing gland.

The 3/8-inch inertia line and volume tank at the top of the loop were replaced with a 15-foot length of 1-1/2-inch pipe. This pipe terminates at a rupture disc located outside the building. The total gas volume of the top section is still 100%. A new tube bundle was installed in the heat exchanger with modifications made to the center rupture tube to provide more rigid construction and, thus, a more effective seal. A new gas drier bubbling system was fabricated and installed at the test site to dry the nitrogen gas used during testing.

SODIUM-WATER REACTION TEST

A series of tests is planned on a seven-tube, bayonet-type bundle so that Na-H_2O reaction information can be obtained for this type of steam generator. The design of the tubes and the shell, which will accommodate a 17-tube bundle, has begun. The initial Phase-II NaK-water tests on small diameter tube bundles were made on the basis that the shell and liquid metal piping be designed to withstand the "closed" system reaction pressure; the new "open" system test equipment will incorporate a 24" diameter low-pressure relief diaphragm on the shell in an attempt to limit the shell and sodium piping design requirements to relatively low pressure values. Since the relief system may result in 200 pounds of sodium mixed with water being ejected into the test pit, exploratory tests are being made in the pit to determine whether quantity discharges of reaction products can be handled without undue danger or damage to the equipment. These tests consist of pouring molten sodium at 600-900 F into water in the bottom of the pit and recording, by photography and inspection, the results of the reaction. Three exploratory tests have been completed as follows: (1) 15 pounds of sodium at 800 F, (2) 38 pounds of sodium at 745 F, and (3) 50 pounds of sodium at 740 F. Photographic results are not completed, but inspections indicate that enough data was obtained to assure proper safety precuations for the tube-bundle tests.

STEAM GENERATOR TEST

Introduction - The primary purpose of the steam generator test is to prove the feasibility of a once-through, shell-and-tube steam generator having a liquid-metal heat source for use in a nuclear power plant. Erection of equipment for this test was completed at the APDA test site in Detroit Edison's Sibley Quarry in August, 1956, and steam was first generated the following month. The steam generator under test, having water and steam inside the tubes and NaK in the shell, is a hairpin-type, 1000 kw heat exchanger containing a bundle of 7 stainless tubes, each of which is approximately 0.5 inch O.D. and 48 feet long. A description of the equipment employed in this test and testing program has been published as ASME Paper No. 55-A-189, "Test Of A Once-Through Steam Generator With A Liquid Metal As A Heat Source." Steady state tests at 25%, 50%, 75%, 100%, and 120% full load were completed prior to April, 1957.

Transient Tests - A series of transient tests was run in which load changes of from 5% to 30% during time intervals of from 5 seconds to 3 minutes were imposed on the steam generator. After a series of isothermal tests were made (run with various feedwater and NaK inlet temperatures and with the steam space flooded) to check the accuracy of the thermocouples in the steam generator, two additional series of transient tests were made. In the first series, load changes of from 25% to 75% were imposed on the steam generator in 30 seconds; in the second series, load changes of 10% were imposed in 2 seconds. Control response and recovery times during the three series of transient tests are being evaluated. During April, May, and June there were 373.1, 284, and 350.4 hours of operation and 192, 220, and 153.6 hours of scheduled shut-down, respectively. Based on a 5-day week, plant factors of 70.7%, 56.4%, and 69.5% were obtained.

General - The development of a method to determine, with a plugging meter, the concentration of dissolved oxides in the NaK system is under way.

Considerable difficulty has been encountered due to oil in the instrument and control air; erratic operation of some of the control system resulted.

Vertical, Bayonet-Tube Steam Generator - A contract has been let for the detail design and fabrication of a vertical, seven-tube, bayonet-type steam generator that is similar to the reference design steam generator for the Enrico Fermi Plant. This unit will replace the existing horizontal steam generator.

COMPATIBILITY TEST OF BULK SHIELDING

The second series of compatibility tests of carbon and graphite blocks with molten sodium was completed. The specimens were weighed, subjected to outgassing, reweighed, and then "cooked" in 900 F sodium for about 4 hours followed by 4 days at 400 F. The samples were then reweighed to determine sodium absorption and are now undergoing chemical analysis and strength tests.

HANDLING MECHANISM TEST

All major equipment for testing cables and chains alternately in hot sodium and inert gas has been received and is being erected at the APDA Test Site at Sibley Quarry. Fabrication of minor parts and the installation of instruments and controls is in progress. In this test a stainless steel cable and pulley mechanism, such as might be used in a mechanical handling device, will be operated in a tank containing molten sodium, sodium vapor, and inert gas. The mechanism, which is motor driven, will support a heavy cylindrical weight. Automatic controls will govern the raising, lowering, and stopping of the weight for set periods of time.

OIL SUBSTITUTE TEST

A series of tests was run to evaluate the feasibility of Hooker Electrochemical MO-10 Fluorolube, (trifluorvinyl chloride) as an oil substitute in certain sodium systems applications. No reaction occurred when MO-10 was added dropwise at various rates to sodium at temperatures between 90 F and 1610 F, with or without agitation; however, when 5-cc or 10-cc batch portions of this material were added to sodium at 500 F, vigorous reactions were noted. By analysis of the combustion gases for trifluorvinyl chloride vapors, the approximate decomposition or reaction temperature of MO-10 was found to be 370 F. Since contemplated uses of MO-10 would result in minute additions of it to hot sodium, this investigation is being continued.

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