PRDC-TR-53

Monthly Technical Report

Report Period November 1961

Power Reactor Development Company
Contract No. AT(11-1)-476

DISCLAIMER

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Distribution

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SUMMARY REPORT - APDA ACTIVITIES
NOVEMBER 1961

CORE DESIGN

Thermal Conductivity of Shielding - Two series of thermal conductivity tests were run on shielding materials for the Fermi reactor.

The first tests were run to determine whether deterioration, from solid to powder, of the borated graphite beneath the reactor vessel would lower the thermal conductivity sufficiently to result in overheating of the structural material in that region. Samples of solid and hand-ground borated graphite similar to that beneath the vessel were taken from more accessible locations at the periphery of the reactor vessel.

The tests were run by Professor C. F. Bonilla at Columbia University. The samples were placed in pairs of cylindrical, stainless-steel cans, 7 inches in diameter and 1/2 inch high. The test assembly was a sandwich arrangement with a flat electric heater in the center, and successive layers on each side consisting of insulation, test can, insulation, and face plate. The face plates were bolted together and the edges of the assembly insulated. The thermal conductivity was calculated from the dimensions of the system, the power input to the heater, and thermocouple measurements of the temperature at the center of each face of the canned sample.

Tests on 15 samples for various locations and compactions showed mean values of k ranging from 1.46 Btu/hr-ft-F at 500 F to 1.67 at 1000 F, with a standard deviation of $\pm$ 0.24. These values were obtained under slight compression. At zero compression, k might have a lower limit value of 0.74 Btu/hr-ft-F.

The second set of tests was performed to evaluate powdered boron carbide as a possible shielding material. The results on five samples of an average particle size of 175 microns and various compactions, tested in the same manner as the previous set, showed average values of the thermal conductivity ranging from 0.37 Btu/hr-ft-F at 500 F to 0.45 at 1000 F. The data as yet have not been used to evaluate a specific design.

Pressure Distribution - The head available for the subassembly in the Core B arrangement was estimated and presented in Memo CA-1127, along with current estimates of the pressure drop across a subassembly. Briefly, the available head is 109.8 psi, while the head required is 110.1, leaving a 0.3 psi pressure deficit.

Core B Annulus Flow Test - While a final design for the seal strips has not yet been established, the proposed methods of attachment (spot-welding and brazing) have proved satisfactory.

A series of tests is being performed to determine the flow-pressure drop characteristics of the annulus as a function of the angular orientation of the seal strips. Three seal strip angles will be tested in a staggered arrangement.

Hydraulic Test Loop - The flow distribution among the channels of the Core B subassembly is a major component in establishing the hot channel factors and, hence, the feasibility of the design. The most practical method of establishing flow distribution is to use small channel taps in a prototype fuel bundle. Experience with the present Quarry water loop, which was constructed a few years ago as a temporary facility, shows that the uncontrollable dirt level of this loop is too high to permit use of the loop as a test facility when channel taps are used: the channel taps would be blocked quickly. A prototype fuel bundle with channel taps has been constructed. A new stainless-steel hydraulic loop is to be constructed to provide a test facility for evaluating and adjusting the flow among the Core B coolant passages.

The pump characteristics will insure the duplication of conditions anticipated in the Fermi plant when it is operating at full design power (430 Mwt), i.e., 350 gpm at a head of 450 feet.

Detail design and purchasing of large components will be carried on in the next few weeks.

Flat Plate Δ P Data - Surface roughness measurements were made on a dummy MTR fuel plate using a "Proficorder" and "Profilometer". These measurements indicate a surface roughness of $\sim$8-10μ RMS.

Fuel Accountability - APDA representatives attended a meeting which was called by PRDC to resolve the apparent increase in enrichment of the Core A fuel pins. The average enrichment of the special nuclear material which PRDC leased from the AEC was 25.591. After converting the UF_6 to derby metal, Davison Chemical shipped the material to the fuel fabricator at an average enrichment of 25.595 $\pm$ 0.008. Isotopic analyses of samples from the first 72 ingots produced by D.E. Makepeace Co. indicated an average enrichment of 25.67 $\pm$ 0.03. This difference is statistically significant at the 95% confidence level. (For accountability purposes, these 72 ingots are termed Core A-I loading.) The results of the isotopic analyses of the next 61 ingots indicate an average value of 25.75. Upon analyzing this data, National Lead Company of Ohio concluded that the upward trend in enrichment is statistically significant and that it merits further investigation.

The isotopic analyses have been performed for PRDC by New Brunswick Laboratory (NBL), using a neutron activation procedure. NBL indicated verbally to PRDC that they have reason to believe their standards have drifted, causing a bias in their analyses. NBL composited the samples from the 61 ingots that had been analyzed by neutron activation and checked the results by mass spectrometer. The results indicated the neutron activation analyses are biased. Thus, a correction factor must be applied to the isotopic results from the 61 ingots.

Since NBL is not certain when their standards began to drift, it will be necessary to recheck all isotopic results from the Core A-I loading. In addition to this, it is planned to set up a bias check program for future isotopic analyses.

DUMMY SUBASSEMBLY CLEANING

Because current problems and the proposed modification of the filter subassemblies have necessitated the cleaning of dummy subassemblies and other reactor components, PRDC has requested a cleaning procedure that permits reuse of undamaged components. The three primary cleaning methods considered are evaporative cleaning, alcohol cleaning, and steam cleaning.

From discussions of this problem with APDA personnel and with others familiar with sodium systems, it is apparent that delicate reactor components are not returned to sodium service as a general practice. In the case of the dummy subassemblies, APDA did not intend that they would be taken in and out of the primary system routinely.

All three of the cleaning methods being studied may leave traces of sodium hydroxide on the reactor components, especially in tight-fitting regions. It is also known that alcohol cleaning leaves a gummy residue, which is probably sodium alcoholate. The presence of sodium hydroxide can cause stress corrosion cracking, while the gummy residue would introduce additional crud into the reactor. To remove these residues from the components, it was recommended that a rinsing step be added to the cleaning procedures.

Examination of filter subassemblies for in-reactor condition is an additional requirement for most of the subassemblies now being taken from the reactor. Of the three cleaning methods considered, properly conducted evaporative cleaning is the only method that leaves evidence of damage intact.

A letter is being prepared summarizing APDA's cleaning experience and recommending cleaning procedures. The procedure to be used for cleaning is dependent upon the purpose for which the subassemblies are removed for examination.

GRAPHITE

BMI Oxidation Program - The experimental work on the oxidation program, which was conducted at BMI, has been completed. A summary memorandum has been issued analyzing the data that will be included in a BMI summary report not yet issued. A brief discussion of the results of this program is given below.

1. Dry Air

   ANL 3 w/o borated material may exhibit as much as a 60% loss in strength when the sample has lost $\sim$10 w/o. The average weight loss rate (averaged over the duration of the test) appears to be higher at small total weight losses and tends to be about constant after 5% total weight loss. The initial tests are believed to have been in error, because of problems in maintaining test temperatures. Although the initial results indicated a weight loss rate of 0.2 to 0.5%/hr at 1000 F, the new data indicates that the weight loss rate is between 0.05 and 0.14%/hr. At 800 F the weight loss rate has been found to be 0.004%/hr.

   TSX material, which is unborated, has shown decreases in strength as high as 40% for total weight losses up to 10%. The average weight loss rate appears to be lower for small total weight losses of less than 5% and tends to increase and to be constant at larger total weight losses. Initial tests are also believed to be in error; the newer and more reliable data indicates that at 1000 F the average weight loss rate is 0.1 to 0.18%/hr and at 800 F the rate is 0.004%/hr.

Two types of borated (4 to 4.7 w/o boron as B_4C) Speer Carbon Company graphite were tested. During fabrication, one of the samples had been heated to only 1400 F while the other had been heated to 4892 F. There appeared to be some directional effects on the material fired at the higher temperature. The poorer directional results show a decrease of 50% in compressive strength at 10% weight loss. The material fired at the lower temperature showed no directional effects and a decrease in strength of 70% at 10% weight loss when tested at 1000 F. The average weight loss rate for the material fired at 4892 F was $\sim$0.09%/hr compared to values ranging from 1.9 to 4.1%/hr for the material fired at 1400 F. Hence, graphitization (fabrication) temperature appears to have a large effect upon weight loss rate. Three small cermet samples of graphite material supplied by the Ford Motor Company and containing 10, 20, and 30 w/o Fe were tested at 1000 F. The strength decreased more than 58% in all materials for about a 10% total weight loss; however, the final strength was still $>$ 9500 psi in all cases. The average weight loss rate was 1.88, 1.27, and 0.66%/hr for the 10, 20, and 30 w/o Fe samples, respectively.

2. <u>Wet Argon</u> (150 $\pm$ 3 F Dewpoint)

ANL material tested in wet argon saturated at 150 $\pm$ 3 F for 240 hrs at 500 F, and for 48 hrs at 800 F and 1000 F, exhibited average weight loss rates of 1.46×10^{-4}, 2.7×10^{-3} and 7.2×10^{-3}%/hr, respectively. There was also little change in strength. Analysis of boron content after testing indicated that there were no significant boron losses. Carborundum 5 w/o borated material tested at 800 F and 1000 F showed strength losses of 46% and 58%, respectively. There was a loss of 0.41 w/o boron at 800 F and a loss of 1.01% boron at 1000 F. Average weight loss rates were 5.4×10^{-3} and 9.8×10^{-2}%/hr at 800 F and 1000 F, respectively. A sample of Carborundum graphite that was removed from the PST and designated as "hard", compared to the other Carborundum graphite after the 1000 F plant operation, was tested for 48 hours at 800 F. A weight loss rate of 2.3×10^{-1}%/hr was observed with a resultant disintegration (complete strength loss) after 11% weight loss. Boron content decreased $\sim$1.4 w/o.

3. <u>Special Tests</u>

A. <u>Flow Rate Effect</u> - ANL material was tested at varying flow rates in dry air, wet air (150 F), and wet argon (150 F) at 1000 F. There did not appear to be a significant flow rate effect between 20 and ~400 fpm for an eight-hour test although considerable scatter was noted. The average oxidation rates in dry air, wet air, and wet argon were 0.13, 0.164, and 0.011%/hr, respectively.

B. Sample Size Effect - A 3-inch cube of ANL material was tested at 1000 F in dry air to ascertain whether there was a sample size effect. Comparison of weight loss with time for this sample, with the standard sample, 1 inch long x 3/8 inch in diameter, showed only a slight decrease in the weight loss rate. The block lost 10 w/o in 380 hours compared to a nominal-sized sample, which lost 10 w/o in 290 hours. It was concluded that size had only a small effect on weight loss rate.

Moisture Content - The tests on the removal of moisture from the Great Lakes graphite under vacuum at 500 F, 1000 F, and 1500 F were completed. The results show that the nonboronated material had released ~98% of its contained moisture in the 80 F to 500 F range and ~2% in the 500 F to 1000 F range. (There was no significant release in the 1000 F to 1500 F range.) The boronated material had released ~85% to 90% of the moisture in the 80 F to 500 F range, ~14% in the 500 F to 1000 F range and ~1% in the 1000 F to 1500 F range. The data indicate that most of the moisture is released within a 24-hour period.

Samples of EBR-II graphite (a fully graphitized product manufactured by National Carbon Company) were sent to ORNL for determination of the moisture content. The samples will be measured with an apparatus that can detect moisture content of $\lesssim$ 0.001% by weight.

Graphite Heat Transfer - Calculations were made previously on the temperature distribution in the primary shield tank (see July report). The results of the calculations are quite sensitive to the values used for the emissivity of the materials since the majority of heat transfer is by radiation. The calculated maximum temperature is particularly dependent upon the emissivity of the leak detector skirt (carbon steel), the reactor vessel, and the 6 inches of borated graphite next to the vessel. Because the emissivity of each of the materials in the reactor is not known, the values used in the previous calculations were taken from the literature.

Because a range of emissivity values is reported in the literature, additional calculations were made during this report period. The lowest values reported in the literature were used for these new calculations, which showed the maximum temperature in the borated graphite to be 1200 F, as compared to the value of 965 F obtained previously. Oxidation tests at BMI have shown that the oxidation rate increases by approximately a factor of 10 for each 100-degree increase as the temperature approaches 1000 F. As a result of these calculations, it is planned to measure the emissivity of the materials in question.

Plug Graphite Irradiation Test - The probe containing the irradiated graphite samples was removed from the pool of the Ford Nuclear Reactor at the University of Michigan. The unit was dismantled and the radioactive portion stored in a "wall tube" for further decay before final disposal.

MATERIALS

U-10 w/o Mo Irradiation Program - Dimensional and density changes were measured at Battelle Memorial Institute for four irradiated capsules. Each contained six U-10 w/o Mo samples 1-1/2 inches long x 0.158 inches in diameter. Three samples in each capsule were initially in the gamma condition and three in the partially transformed alpha + delta + gamma condition.

In comparing these dimensional and density changes the following observations may be made:

1. There appears to be a correlation between the initial phases present in regard to swelling (measured by density change). In all cases it appears that swelling is greater for the initial "$\alpha + \delta + \gamma$" condition.

2. U-10 w/o Mo operating at an irradiation temperature of $\sim$850 F and lower and a fission rate of $> 2.5 \times 10^{13}$ F/cc/sec is much more stable dimensionally than materials irradiated at $>$940 F and at a fission rate of between 4.2 and 5.0×10^{13} F/cc/sec. Possibly this is simply a temperature effect, but not in view of the critical fission rate idea which has been formulated to explain previous CP-5 and MTR results. It is postulated that all samples operating in the latter (larger swelling) condition are in the transformed state, which swells more readily, while all samples irradiated in the former condition are in the gamma condition. The phases will be checked with metallographic work.

Nonfissionable Materials Irradiation Program - Tensile samples of Inconel-X were received from the International Nickel Company. These samples are fabricated to APDA specified dimensions from four different types of heat treatments: singly-aged; doubly-aged; solution-treated; and a special heat treatment, which INCO claims will give better properties. Twenty-five samples of each heat-treatment were provided. It is planned to irradiate these materials along with type 304, 347, and 316 SS already fabricated for surveillance work.

The Inconel-X samples will be inspected by APDA inspectors in the next month.

NUCLEAR ENGINEERING

Graphite Shield Design - A memorandum was issued on the axial heating distribution in the inner and outer borated layers of the graphite shield for both Core A and Core B operation. In this memorandum the midplane heating values are revised somewhat from the values previously obtained.

Another memorandum was issued which gives the 20-year fast nvt's and helium generation in the midplane of the graphite shield for Core A and Core B operation.

A memorandum was also issued which gives for Core B and Core C operation the neutron flux levels, heating, and 20-year nvt's in the lower head region of the graphite shield underneath the vessel. The heating in the 6-inch borated graphite layer here is about the same for both Core B and C, and it is about 100 times smaller than the corresponding heating in the radial inner 6-inch borated graphite layer. Neutron and gamma fluxes on the lower structural concrete support pad were also determined. It was found that the neutron flux on the concrete is increased by a factor of about 10^4 if all of the boron in this area is lost due to a chemical reaction with water vapor.

Work is proceeding on the nuclear heating in the flex legs and the graphite shield redesign in the area between the vessel and transfer rotor container. Problems have been run for both of these studies, and the results are being analyzed.

Accuracy of Shield Calculations - A comparison was made between the neutron production in the radial blanket as calculated by the AIM-6 code and as determined from critical experiment data. For both Core A and Core B, the calculated production is lower than the experimentally determined value. The discrepancy becomes progressively larger with distance into the blanket. Since the neutron leakage to the graphite shield is dependent on this value, thought is now being given to the correction factors that should be applied to the AIM-6 shield calculations because of this effect.

Recent AIM-6 calculations of the neutron attenuation through sodium and plain graphite seemed to indicate much larger attenuations through these materials than were obtained by previous straight-ahead hand calculations. This problem was investigated and it was found that no actual discrepancy exists between the two methods. The difference can be accounted for by the different spectrum of the source neutrons used in each case. During the course of this study, however, a comparison was also made between calculated neutron attenuations and the limited experimental data on neutron attenuation available. The comparison showed that the discrepancy between calculated and measured response of a BF_3 detector becomes worse the further into the shield one goes. Since the calculated response is smaller than the measured response, it was concluded that calculated boron heating values in the graphite shield are probably too low also. When this investigation is completed, the confidence factors now being used in the shield calculations will be reviewed.

An investigation is also being made of the observed discrepancies in the neutron leakages (currents) into shield regions as calculated by the AIM-6 code. In some cases negative leakages are obtained although this is physically impossible. The trouble appears to be in the way the code handles the leakage calculation numerically. The method used does not apply well in shield regions where the flux is low; however, no inaccuracy exists in the flux calculations, upon which the heating and flux levels throughout the shield are based. Although the discrepancies occur because of the way the code handles these fluxes numerically to obtain leakage values, a method of correcting the discrepancies is being developed so that the results will be consistent, and as another check on the calculations.

General Shielding and Health Physics Studies - A memorandum was issued on the graphite moderator requirements needed around a Pu-Be neutron source located in one of the detector tubes. The source is needed to test out the neutron detectors during the preoperational test program.

A study was also completed on the allowable maximum leak rates of sodium-24 vapor, fission product gases, and argon-41 into the sodium control room, in event of a sodium spill in the cold trap room.

Calculations were made of the neutron and gamma shielding requirements for the cover gas fission product detectors located outside the containment building. The shield consists of 30 inches of concrete.

Safety - Work continued on the below-floor problems associated with the hypothetical accident case. Discussions with Ray Brittan of ANL, staff members of Stanford Research Institute (SRI), and others have produced no definite mechanism for relieving the conditions that cause the problem. It has been suggested that refinements in the calculations such as the inclusion of conductive and convective heat transfer to the sodium, more realistic boundary conditions on the vessel, and others may prevent a calculated vessel rupture. SRI has suggested that an analysis of energy partition in some simple experiments may provide a theoretical explanation of the low amount of energy which appears as mechanical work in many experiments. Also, they suggested that a shock-wave analysis be made for the accident being considered. This will provide energy dissipation as heat in sodium and structural material. However, no action was taken in any of these directions. Rather, more thought is being given to the entire problem. As an aid, a parameter study is being made on a simple model in which the only form of energy dissipation is the movement of the plug. These calculations will point out those properties of the system that are critical and deserve further attention.

Nuclear Test Program - Preliminary drafts of nine nuclear test procedures for the Fermi plant were completed during the month. These drafts will be submitted to the PRDC operating staff for comments on completeness and accuracy of operational procedures before final drafts are submitted for technical approval. The test procedures have been prepared according to a general format established by the review of procedures prepared for other reactor facilities. The following nuclear test procedures are in preliminary draft form:

1. U-235 Worth, Shim Subassemblies

2. Edge Fuel Worth

3. Flow Dependence of Reactivity

4. Isothermal Temperature Coefficient

5. Flow Coefficient Tests at Power

6. General Reactivity Measurements

7. Preliminary Isothermal Temperature Coefficient

8. Power Coefficient

9. Control Rod Calibration

A number of preoperational tests in progress at the Fermi plant are of direct interest to the nuclear test program. Personnel of the Nuclear Test Group have been assigned to follow these tests to assure that sufficient data is obtained to satisfy the requirements of subsequent nuclear testing and operation. Current preoperational tests include bench testing of the neutron detectors for start-up. This testing is currently being held up by the delivery of auxiliary neutron sources. A neutron moderator geometry was fabricated during the month for use with the sources. The British Transfer Function Analyzer has been activated for checkout and bench testing. A number of component failures have been uncovered, and British replacement parts are required before testing can continue. The steam and feedwater control system has been interconnected with the plant simulator, and preoperational tests are scheduled to begin in December as a portion of the plant systems verification program. The preoperational tests of the reactor safety system have been terminated because of numerous design inadequacies that must be rectified before the preoperational tests can continue. A memorandum was issued requesting an accurate determination of the available void volume in the sodium-void fuel subassemblies for in-core physics measurements. Design considerations resulted in the decision to utilize a gas volumetric measurement. A recommendation for such a measurement along with a recommended test procedure will be forwarded to PRDC.

The foil irradiation requirements of the Fermi shield test program were included in the in-core foil irradiation program, to consolidate all foil requirements and counting and handling equipment. A linear amplifier-pulse height analyzer borrowed from Hamner Electronics was operated with available scintillation-counting equipment as part of a continuing program to evaluate new commercially available components for application in the foil irradiation program.

The Marquardt Corporation was consulted regarding the characteristics and availability of a special-purpose high-temperature neutron detector which was developed for the nuclear ram-jet system. Unfortunately, the detector characteristics were not compatible with our requirements for an in-core start-up detector.

A presentation was given before the PRDC Safety Committee on the progress of the EBR-II and Dounreay Fast Reactor experimental programs. The reactor operator training sessions were continued in preparation for the operator licensing examinations.

Core B Critical Experiment - A number of miscellaneous items were accomplished during the month: (1) The measurements of material worths were completed for a great number of elements throughout the entire core. In general, the agreement between measured and calculated values was fairly good, the most disagreement being found for U-238, nickel, and carbon. (2) Some preliminary fission-ratio measurements were made, but the bulk of these measurements will be performed later. (3) The worth of a safety rod as a function of its distance in the core was measured. A fully inserted rod is worth nearly $2.00 in Core B, approximately twice the corresponding value in Core A.

The AEC has been requested by PRDC to extend the ZPR-III time allotted for the Core B Critical Experiment. The original time requirement did not take into account the fact that a nickel reflector would be required and that other reflector experiments would be needed. These requirements will add approximately 6 weeks to the program, and it is expected that the additional time will be allowed. No conflict with ANL programs exists since they are still awaiting approval of a plutonium core experiment, which is planned to follow the Core B experiment.

Cross Sections - Two new sets of cross sections have been used to determine reactivity for a Core B base case sphere. The first, a revision of the AIM 18-group set in present use, overpredicted reactivity by 3.0%, as compared to 7.5% for the present set. The second, a revision of the 16-group set of Yiftah, Okrent, and Moldauer (YOM), gave a 2.7% overprediction. These more favorable reactivity results indicated the desirability for a complete analysis of danger coefficients, for comparison with experimental results, using both sets. This study, now in progress, will allow a complete checkout of the modified AIM-6 Code (MAIM) with an included option which gives automatic PERT calculations for determination of danger coefficients.

A conference was held with personnel at Argonne National Laboratory regarding methods of obtaining valid multigroup constants for reactor systems containing light resonance scatterers. As a result, parallel calculations are being performed here and at ANL using, respectively, the RESPECT and ELMOE codes. The purpose of the calculations is to determine the effect of the use of the narrow resonance approximation for closely spaced resonances. Efforts to determine spin assignments more exactly from the experimental data of ANL's C. T. Hibdon were mostly unfruitful.

An 18-group inelastic scattering matrix for sodium has been calculated from the experimental data of Freeman and Montague. This matrix will be utilized in AIM-6 calculations to determine effects on reactivity and danger coefficient values.

Programming & Computer Operations - The work accomplished during the report period included the following items:

1. 17 SWAMI problems were run.

2. A number of matrix inversion problems were run in connection with flux perturbations in small samples.

3. Preliminary work was started on fitting equations to a number of curves that will be needed by what will eventually be known as the program HYPO. This program will calculate the energy dissipation in the maximum hypothetical accident and will be an IBM-7090 version of the Allis-Chalmers/Bendix program.

4. A step-wise regression calculation was performed to attempt a fit of the tabular function $J(\xi, \beta)$. Analysis of the results has indicated that the functional form of the fitting equation should be modified. The modification is now underway.

5. Seven "Heating Code" problems were prepared, and four were run successfully.

MECHANICAL HANDLING

Technical Liaison - Some preliminary work was done during the report period on design of equipment for securing the hold-down mechanism in the "up" position to permit disassembly of the hold-down superstructure without lowering the hold-down if disassembly becomes necessary.

Remote Maintenance - Work was continued on modifications to remote maintenance equipment in preparation of removal of the OHM for repair.

ELECTRICAL AND INSTRUMENTATION

Malfunction Analysis - Analyses were made of the Primary Inert Gas System and the Building Ventilation System to determine the consequences of an instrument malfunction and the consequences of a complete loss of supply air to the control systems. The analysis of the Primary Inert Gas System indicated the possibility of dumping radioactive gases into the above-floor containment building upon the loss of air to the control system. As built, the Klozure seal relief valve is vented to the containment building. However, for radioactive gases to be vented to the containment building, gas must bubble through the dip seal and the inner part of the Klozure seal must leak. The potential problem was called to the attention of PRDC, and a work order has been issued to connect the Klozure seal vent line to the dip seal vent line, which in turn vents to the waste gas system.

The results of the analysis of the building ventilation control systems indicate that no serious consequences can develop from loss of control air or from the malfunction of an instrument.

Pre-operational Tests - The detailed pre-operational test procedures for the Null Balance Transfer Function Analyzer were transmitted to PRDC.

Periodic Pressure Tests for Containment Building - Analyses were made of the problems associated with instrumentation and control systems inside the containment building when the building is pressure tested at 2 psi while the reactor is operating. The analysis indicates that it is not feasible to operate the reactor while the building is pressurized; a large number of interlocks must be by-passed and adequate control of several variables is questionable. Work is continuing to determine the modifications to control systems that would permit pressure-testing when the reactor is shut down.

Gas Analyzers - Work continued on defining optimum instrumentation for detecting hydrogen in the secondary system cover gas; detecting H_2, O_2, N_2, CO, CO_2, and CH_4 (methane) in the primary inert gas; and detecting sodium vapor in the primary inert gas downstream of the vapor trap.

Hydrogen detection instrumentation applicable to the secondary system cover gas is commercially available in the ppm range. Chromatography techniques can be readily used to detect hydrogen concentrations of 10 ppm, by volume, in argon. For concentrations below 10 ppm, the instrumentation exhibits rather poor sensitivities. Thermal conductivity devices are available for detecting hydrogen in argon in the range of 0 to 10 ppm, by volume. The units exhibit good sensitivities for 1 ppm. The major problems with the thermal conductivity devices is that they are not selective. For example, the presence of approximately 20 ppm, by volume, of nitrogen in the argon gas would give the same reading as 1 ppm, by volume, of hydrogen.

A recommendation has been made for the installation of three thermal conductivity units, one for each steam generator, plus one chromatograph with provisions for alternately sampling the argon gas for hydrogen from each of the three steam generators. A contact on each of the thermal conductivity units would be set to sound an alarm at a concentration 1 ppm of hydrogen by volume. If desired, the chromatograph could be switched to the cover gas of the particular steam generator that had initiated the alarm. Since the chromatograph is selective, one could determine whether the presence of hydrogen or the presence of nitrogen had caused the alarm. This would differentiate between a sodium-water reaction and leakage of nitrogen (air) into the secondary systems.

The gas chromatography technique is also applicable to the detection of H_2, O_2, N_2, CO, CO_2, and CH_4 in the primary inert gas. This can be accomplished by using a chromatograph with two columns. A proposal for the instrumentation for this application is expected in December.

To date, the search for commercially available instrumentation to detect sodium vapor downstream of the vapor trap has been unsuccessful.

Liquid Metal Level Detector - A conceptual design report describing a level detector for liquid metal applications was released in November. Approval for funds to develop the detector was obtained. The design of the detector will be such that the unit can be installed in the Fermi reactor vessel. Some of the salient features of the detector are as follows: infinite resolution, insensitivity to temperature changes, in-place calibration, extremely fast response, and no moving parts.

LIQUID METAL AND STEAM SYSTEMS

Primary Sodium Pumps - Pumps No. 2 and No. 3 are being operated intermittently at pony motor speed to maintain an even temperature distribution in the system while the hold-down mechanism is raised.

No. 1 pump shaft seal is at Byron Jackson being lapped.

Primary Inert Gas System - Feasibility studies were made to provide a direct purge of the reactor cover gas. Modification of the storage and purification controls were recommended to provide sufficient delivery capacity, a workable reserve capacity, an alarm, and a more reliable reactor building supply. A modification to the line equipment in the holdup and surge tank outlets to extend administrative control was suggested.

Vacuum Distillation Facility - The vacuum distillation program was indefinitely postponed on November 15, and all records, designs, and specifications were transferred to Liquid Metals File 7629 for future reference.

Primary Sodium - At the request of PRDC, a recommendation was made on a sodium transfer procedure to lower the reactor sodium level for visual examination at the loading face of the core. A study was made of cold-trapping procedure at low levels.

A feasibility study will be instituted early in December to provide for removal of carbon from the primary sodium system by hot trapping.

Vacuum Distillation Tank - Thermal calculations of heating coil performance, refractory surfaces, and the sodium condenser were completed.

Sodium Heating System for Hot Trap - Sodium-to-sodium heat exchanger calculations were completed.

Reactor Building Periodic Leak Test - Further study of a periodic 2-psig leak test of the reactor building indicates that it would be unsafe to operate the reactor during the test. Therefore, a detailed procedure is being prepared for conducting the test during a weekend shutdown.

TEST OPERATIONS

Endurance Testing Loop (Water) - The test loop was cleaned internally, and some additional construction work was done in preparation for another phase of the Core B test program. In this phase the distribution of flow through the various channels of a full-scale test unit is to be determined. Further preparatory work has been delayed pending the construction of a new stainless-steel test loop that will improve test conditions.

Endurance Testing Loop (Sodium) - In the final stage of preparing the loop for test operations, the loop was subjected to gas pressure and sodium leak tests. No leakage was detected in the course of (1) a 10-psi gas pressure check, (2) repeated sodium flushing at temperatures between 500 and 800 F, and (3) a 24-hour sodium flow of 300 gpm at 900 F. The validity of the manufacturer's calibration for expected loop conditions was verified, and the test loop is now considered ready for service.

Klozure Seal Test - After the installation of a new outer seal ring, the space between the seals and the gas supply line to the seal assembly were pressure tested again. Excessive rate leakage from the seal assembly was reduced to a point considered acceptable for ambient conditions, by displacing the upper seal plate one-quarter revolution. Ambient measurements of top plate torque for both atmospheric and 1-psi seal pressure were obtained and will be compared with those to be taken at 250 F. Another pressure test of the space between the seal rings will be carried out at the same elevated temperature.

Gross Can Behavior Test - Testing of the plain and borated graphite samples in a 1000 F sodium vapor atmosphere continued through the month. At the end of the month, the samples had been under test for 91 of the intended immersion period of 210 days. Examination of the samples taken from the liquid sodium environment at the end of the previous month yielded the following information:

		Borated	Unborated
Test period	hours	1394	1515
Change in can width (average)	inch	-0.022	+0.064
*Change in can height (average)	inch	+0.041	+0.077
Graphite block width (average)	inches	2.6987	2.7440
Graphite block height (average)	inches	10.500	10.669
Sodium content	%	15.6	16.0
Weight increase	grams	533	536

*The top of both cans had bulged slightly in the center. The measurements indicate the amount of bulge.

Both the borated and unborated graphite protruded through the holes in the top of their respective cans about 1/8 inch; also, both types of blocks had swollen so much that the cans had to be cut apart to permit removal of the blocks. A crack extended the full length and width of the borated block. In the unborated graphite block, smaller cracks extended from the faces into the center, but not completely through, the block. The testing of specimens in liquid sodium is now complete.

Interaction Rate Test - In another series of tests, two plain graphite cylinders, each 1.75 inches in diameter and 2 inches long, were immersed in 1000 F liquid sodium for 72 hours and then removed for examination. Both graphite cylinders had previously been out-gassed at 1000 F for 24 hours. The following changes had occurred:

		Specimen 1	Specimen 2
Change in diameter (average)	inch	+0.1383	+0.1131
Change in length (average)	inch	+0.0850	+0.0686
Change in weight	grams	+22.5482	+29.4445

Samples of the sodium used in the bath and samples of the stainless-steel basket (for holding the specimens) were taken before and after the immersion period; all samples were sent to the Core Analysis Section for subsequent analysis. The purpose of the analysis was to determine (1) whether any graphite was lost to the sodium, and (2) the effect of sodium immersion on the stainless-steel basket material. This analysis completed the interaction rate tests.

Restrained Swelling Test - Testing continued on graphite samples for the determination of swelling and sodium absorption characteristics of the samples when subjected to a 680-pound circumferential restraint and immersed in high-temperature sodium. Tests were conducted on graphite samples under different conditions with results as follows:

Sample	Test Period Hours	Sodium Temperature F	Diameter Increase %	Weight Increase %
Great Lakes Semigraphite - 5% boron:				
25G-4	24	900	1.9	20.1
13G5-9	24	900	3.9	29.0
National Semigraphite (plain)				
AGX-13	24	500	0.59	7.19
Great Lakes Semigraphite - 5% boron				
13G5-12	24	500	2.08	18.21

The above samples are different from those tested previously.

Relaxation Testing of Inconel-X Springs - Testing continued on the dimensionally modified Inconel-X spring to determine its suitability as a subassembly nozzle spring. This modified spring and a normal subassembly spring were compressed by application of 538.5 pounds, heated to 850 F in air, and maintained in this condition for 15 days. At the end of this period the required load for full compression had decreased 110.6 pounds for the normal spring and 105.5 pounds for the modified spring. A subsequent check showed the effect of temperature and loading on the free spring lengths. For the modified spring, the post-test free length was 7.7% less than the before-test measurement; the decrease was 7.3% for the normal spring.

Stuck Subassembly Test - The nozzle sleeve was immersed in 500 F sodium for 500 hours and post-test hardness and diameter measurements taken. These, when compared with before-test measurements, indicated a decrease in hardness of the nitrided sleeve from an average 56.0 to an average 53.1 on the Rockwell "C" scale, and an average decrease in diameter of 0.0001 inch. Actual diameter changes varied between the extremes of +0.0011 inch and -0.0017 inch.

The nozzle spring was artificially corroded by immersion in water at 212 F for 24 hours, during which time air was bubbled through the nozzle. Corroded spring characteristics were then compared with those of a clean "control" spring at ambient temperature and at 500 F in air to determine the effects of spring corrosion. A generalization of results which apply for both the ambient and 500 F conditions follows:

1. Upon application of the load on the corroded spring, the spring moved erratically within the first 0.003 inch compression because the spring stuck within the nozzle sleeve. The clean spring moved freely in this range.

2. Beyond the 0.003 inch point, the corroded spring and the clean spring compressed freely with increasing load.

3. Also beyond the 0.003 inch compression point, both the clean and corroded spring deflected at the same rate throughout the balance of the load range. For the same deflection, however, the corroded spring required 5 pounds less loading, this difference being consistent throughout the deflection range (beyond the 0.003 point).

Core B Plate-Rib Joint Thermal Cycle Test - After assembly of test equipment and installation of a 3-plate unit to simulate the core bundle, the plate-rib joints were dye-checked for evidence of any before-test cracks; two small cracks showed on one side of the plates. The test consisted of cycling the temperature of the plate bundle between 1000 F, which created a temperature differential of 65 F across the face of a monitored plate, and some lower temperature that would create a differential of 10 F. At the end of the first cycle, four apparent cracks were found on the same side of the plates as the before-test cracks. After 23 cycles there were 9 apparent cracks. As a further check to verify the presence of actual cracks, the suspect plate-rib joint was lightly dressed with emery cloth, cleaned with a 15% solution of nitric acid and neutralized with water; then again dye-checked. This time there appeared to be no cracks. Possibly, the cracks detected before the acid cleaning may have been the result of some oxide formation on the metal surface. Testing of the plate bundle will continue through another 27 cycles.

THE SANITATION FOUNDATION

HEADQUARTERS

SCHOOL OF PUBLIC HEALTH • UNIVERSITY OF MICHIGAN

ANN ARBOR, MICHIGAN

January 2, 1962

Mr. Robert W. Hartwell
General Manager
Power Reactor Development Company
1911 First Street
Detroit 26, Michigan

Dear Mr. Hartwell:

The results of the airborne dust sampling and analysis program for PRDC in November, 1961, are as follows:

Location	Sample Number	Date Collected	Radioactivity Content (μc/ml)	Counting Error at 95% Confidence
Reactor Site	1D44A	11/3/61	1.3×10^{-11}	4%
	1D44B	11/10/61	7.2×10^{-12}	6%
	1D44C	11/17/61	1.1×10^{-11}	5%
	1D44D	11/24/61	7.6×10^{-12}	5%
Ann Arbor	61D44A	11/3/61	1.0×10^{-11}	4%
	61D44B	11/10/61	6.0×10^{-12}	6%
	61D44C	11/17/61	1.4×10^{-11}	4%
	61D44D	11/24/61	6.8×10^{-12}	5%

The radioactivity concentrations in air during the month of November remained relatively high with considerable variation from week to week.

Charles A. Pelletier

Charles A. Pelletier, Director
Radiological Health Services

CAP:dml