YAEC-70

YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

MONTHLY PROGRESS REPORT

APRIL, 1958

R&D SUBCONTRACT NO. 1 under
USAEC-YAEC CONTRACT AT (30-3)-222

MAY 20, 1958

WESTINGHOUSE ELECTRIC CORPORATION

ATOMIC POWER DEPARTMENT

PITTSBURGH, 30 P. O. BOX 355 PENNSYLVANIA

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Yankee Atomic Electric Company

Research And Development Program

YAEC-70

MONTHLY PROGRESS REPORT

for the period

April 1st to 30th, 1958

by

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R. W. Garbe

Large Plant Engineering

For The Yankee Atomic Electric Company
Under Research and Development Subcontract
No. 1 of USAEC-YAEC Contract AT(30-3)-222

May 20, 1958

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ABSTRACT

This report describes the work performed or coordinated by the Westinghouse Atomic Power Department for Yankee Atomic Electric Company under Research and Development Subcontract No. 1 of USAEC-YAEC Contract AT(30-3)-222, during the month of April 1958. YAEC Development Program Report, YAEC-41, Rev. 2, outlines the Research and Development Program for the period from January 1 to June 30, 1958.

INTRODUCTION

This report describes the Research and Development performed during April 1958 by the Westinghouse Atomic Power Department for the Yankee Atomic Electric Company as covered in YAEC Contract AT(30-3)-222 with the Atomic Energy Commission. The program, which is detailed in YAEC Development Program Report YAEC-41, Rev. 2, outlines the research and development required to build a 134 MW (net electrical output) pressurized light water nuclear reactor power plant having a core of slightly enriched uranium dioxide (UO_2) fuel pellets contained in stainless steel tubes.

Quarterly Progress Reports, YAEC-7, YAEC-13 (Revision-1), YAEC-20, YAEC-35, YAEC-44, YAEC-52 and YAEC-65 describe and evaluate the work accomplished from the beginning of the program, June 6, 1956 to March 31, 1958.

1.0 FUEL ELEMENT DEVELOPMENT

The work under this project is directed toward developing a satisfactory stainless steel clad UO_2 fuel element and is divided into the following subprojects:

1.1 Uranium Dioxide Fuel Material Preparation

The proposed UO_2 powder specification was coordinated with the UO_2 vendors and conclusions are now being integrated into a final specification. A procedure for sampling and quality control of the UO_2 powder to pellets was prepared and issued.

In connection with the ceramics development program at the Westinghouse Materials Manufacturing Department, a simplified granulation procedure which reduces the number of pre-pressing operations on pellets has been developed. Trial pellets have been made using this procedure prior to pressing and sintering. If the sintered pellets are satisfactory regarding dimensions, mechanical and chemical properties, full scale production trials will be conducted using this greatly abbreviated procedure.

1.3.1 End Closure of Fuel Rods

A welding procedure has been written which specifies physical settings, cleanliness requirements, and periodic checks required for proper functioning of the welding equipment.

1.3.2 Joining Fuel Bundles Into Assemblies

A stretch-formed tube, in which discs separating the compartments were brazed in position by passing sections of the tube through the hot zone of the brazing furnace, was sectioned and each compartment checked for leak tightness using helium leak test techniques. Seven of the eight discs in the tube were brazed leak tight. Each disc was heated at a temperature of 1875°F for one-half hour during the brazing treatment.

A second prototype Yankee sub-assembly fuel element was brazed at the Ferrotherm Company and inspected. The maximum bow was approximately .055". The minimum spacing between fuel tubes was .035" (design is .088").

Brazed joints strength studies are being conducted using the three most promising methods of braze application which are Nicrobraz 50 slurry, pre-plated NiP .001" thick, and combination of pre-plating with .0005" of NiP and back filling with Nicrobraz 50 slurry.

Joints with fracture loads of approximately 300 to 800 lbs have been reached with heat treatments of 30 minutes to one hour at a temperature of 1850 to 1900°F.

1.4 Fabrication and Analysis of Fuel Elements for Critical Assembly and Irradiation Tests.

A decision was reached at a meeting on April 25 with the representatives of the Yankee Atomic Electric Company on a program describing the specimens for the MTR in-process water and in-pile loop irradiation experiments. Approval of this program is being awaited.

A heat transfer analysis was conducted on 30 inch long specimens for the MTR process water irradiations. These specimens can be used at the required flux level without exceeding heat transfer limitations.

2.0 NUCLEAR DESIGN AND REACTOR PHYSICS

This project includes study and calculations of the reactor core, criticality experiment, irradiation experiment, shielding, and the reactor startup and operation.

2.1 Core Design Optimization

No work was performed under this subproject during the month of April 1958.

2.2 Core Steady State Analysis

Two region cycled cores of 5000-5000, 7000-7000 and 10000-10000, hour cycling were examined. Little, if any, improvement in radial hot channel factor was noted over a uniform core. Non-cycled cores with two regions of different sizes and enrichments have also been studied which have a 30% improvement in radial hot channel heat flux factor.

The two-dimensional diffusion theory computer program (PDQ) was checked out on the IBM-704 computer. This program will provide a means for determining neutron flux distribution in a cylindrical geometry.

A generalized geometrical representation of the first core has been set up for use with the PDQ program. It will permit the study of cores of various loading arrangements and control rod configuration with minimum duplication of effort in preparation of input data.

2.3 Core Kinetic Analysis

The calculation of temperature coefficients of the first core was completed. A topical report "Temperature Coefficients of the First Yankee Core", is being prepared which will present this work.

A simulation procedure was decided upon for use in the next analog computer studies on the startup transients in the first core. This procedure will incorporate techniques developed in the complete loss of flow study and in recent analysis of transients associated with critical experiments.

2.4 Control Rod and Chemical Poison Anslysis

An evaluation of the control rod program for the first core is continuing.

2.5 Critical Experiment - Design and Analysis

The JOFIT code, for least squares fitting of radial flux plots to obtain bucklings, became operational.

Preliminary temperature coefficient experimental data obtained from the 3:1 CRX core were in good agreement with previously calculated values. Temperature coefficients were calculated for a water-uranium metal ratio of 4:1.

The CANDLE 4-group burnout code with SOFOCATE thermal constants gave a k_{eff} of 1.017 for the just critical cylindrical configuration of 1862 CRX fuel rods.

In connection with the hazards analysis of the Yankee critical experiments, studies of the transients associated with uncontrolled rod withdrawal, poison dilution, and water insertion were made for the 4:1 and 3:1 water-uranium metal ratio experiments in addition to those for the 2.23:1 case. A check solution was also run on the IBM-704 computer by means of the STIRRUP Program. The STIRRUP Program was also used in the analysis of the maximum credible accident. The hazards report for the 2.23:1 CRX core was completed and issued as Supplement 1 to YAEC-31, "Yankee Critical Experiments, Hazards Summary Report".

2.6 Irradiation Experiment - Design and Analysis

Calculations were performed to determine into which MTR test hole location new fuel samples should be inserted so that a heat flux of 600,000 Btu/ft^2-hr would be obtained.

2.7 Shielding Analysis

No work was performed under this subproject during the month of April 1958.

2.8 Startup Experiment Assistance

No work was performed under this subproject during the month of April 1958.

3.0 CHEMISTRY

The effort on this project is directed toward establishing methods of utilizing chemical poisoning for reactor control and studying the crud and corrosion problems in the reference environment.

3.1 Properties And Removal of Chemical Neutron Absorbers

Tests were conducted in comparing the relative behavior of Rohm & Haas XE-150 (mixed-bed neutral resin) and XE-170 (mixed-bed elevated pH resin). The decontamination factors in the higher pH levels are reduced compared with those obtained at the lower pH levels and this is believed to be due to the formation of radiocolloids at the higher pH conditions.

3.3 Corrosion of Materials of Construction

Test No. 5 in Ⓦ APD PAR Loop "A" which was a screening test of materials to be used in the Yankee Reactor in water with boron less than 3 ppm was completed during this period. The duration of the test was 707 hours at a temperature of 600°F, pressure of 1831 psi, flow velocity of 37.7 fps, hydrogen overpressure of 31.2 ml (STP)/Kg solution and pH of 6.1. In general, there is a decrease in extent of corrosion with decreasing boron concentration as compared to the tests previously reported which were run with a higher boron content. An anomalous result has been obtained with Ag-In-Cd alloy in this test where a marked weight increase was obtained as compared with the weight increase previously obtained in tests with higher boron concentrations. A specimen of the Ag-In-Cd exposed in this test is now being examined metallographically.

A temperature cycling test (cycle 150° to 600°F) was run on a brazed fuel bundle in a dynamic autoclave (.1 fps) containing neutral degassed water with a hydrogen over-pressure sufficient to give 30 cc H_2/Kg H_2O at STP. After 14 cycles, the bundle appeared unchanged from its original condition.

Corrosion testing of Ag-In-Cd alloy in 600°F water containing 1600 ppm of boron and pH adjusted to $5.8 \pm .2$ with KOH indicated an initial weight gain of 15.5 mg/dm^2 for a three-day period.

3.4 Interactions Between Chemical Absorbers, Corrosion Products and Fission Products

This project supports a study of the effects of Van de Graaff electron beam irradiation on crud deposition at both neutral and elevated pH. Starting difficulties and insufficient heating capacity of the loop delayed the test during this period. It is expected that the tests will begin during May.

3.5 Decontamination And Waste Disposal Studies

An experimental test program using cleanup agents manufactured by the West Chemical Corporation of Long Island City, New York was undertaken. Preliminary results indicate that "WEDAK", a detergent containing highly inhibited phosphoric acid, is capable of dissolving Yankee type crud.

A report on "The Possibility of Caustic Embrittlement in Basic Permanganate Decontamination" was completed. The conclusion reached on the basis of a literature survey and experimental test program is that there is no significant possibility of the occurrence of either "caustic cracking" or "embrittlement" of stainless steel as a direct result of following the basic permanganate-citrate decontamination procedure.

3.6 Crud Inhibition, Suspension and Removal

Engineering was begun in this period of a crud deposition loop which will measure effect of various parameters on crud deposition on a heat transfer surface when heat flow is from the coolant to the solid container (a simulated steam generator condition). This crud deposition loop will operate in a side stream off Ⓦ APD PAR Loop "A".

4.0 MECHANICAL DESIGN

This project includes the design and development of mechanical features of fuel assemblies, control rods, baffles, the support structure, the reactor vessel closure and fuel handling tools.

4.1 Fuel Assemblies and Control Rod Design

The model of the proposed spring-loaded control rod latch was completed and tested. Although the mechanism worked satisfactorily, a few revisions will be made to improve the design.

4.2 Control Rod Drive Mechanism

The Westinghouse Atomic Equipment Department has completed all the detail drawings of the prototype of the positive engagement type control rod drive mechanism and has released them for material purchasing.

The cooling requirements for the control rod drive mechanism were calculated.

4.3 Design Of Core Support Structure And Fuel Handling Tools

A preliminary design of the universal fuel assembly and control rod handling tool was developed. The design study is being continued.

4.4 Design For Critical Experiment And Irradiation Tests

No work was performed under this subproject during the month of April 1958.

5.0 THERMAL AND HYDRAULIC DESIGN

This project is directed toward the development of a design which will have satisfactory thermal and hydraulic characteristics under conditions of steady state, transient, and emergency operating conditions.

5.1 Thermal Design

A study was begun to determine the effect that differential expansion will have on the tension stresses and compression bowing of the fuel tubes in fuel sub-assemblies.

5.2 Hydraulic Design

An analytical method was devised to obtain the volume of the first core in local boiling as a replacement of the graphical method. Calculations are being made to determine the conditions caused by the failure of two pumps.

6.0 CONTROL ROD DEVELOPMENT

This project involves the development of designs and specifications for reactor control rod material.

6.0 Control Rod Development

Grain size determinations were made on the as-extruded shapes of Ag-In-Cd control rod material and the results show that the grain size was too fine to achieve the maximum creep strength for the material. Samples have been obtained and some of these were subjected to various heat treatments to determine the most suitable temperature and time for annealing the as-extruded cruciforms to result in a grain growth which will give an acceptable grain size .

7.0 INSTRUMENTATION AND CONTROL

This project covers the investigation and development of an overall control system and instrumentation including analyses of system functions and development of specifications for system components.

7.0 Instrumentation and Control

No work was performed under this project during the month of April 1958.

8.0 PLANTS SYSTEMS DEVELOPMENT

This project involves the analysis, evaluation and development of plant systems including primary coolant, make-up and purification systems for the contemplated reactor.

8.5 Chemical Handling And Control Systems

No work was performed under this subproject during the month of April 1958.

8.11 Reactor Handling Tools And Plant Shielding Analysis

The conceptual design of the plate and barrel handling fixture was completed.

Design effort was continued on the plate and barrel handling mechanism mock-up for testing in the Ⓦ APD High Bay Building Deep Pit.

9.0 PLANT SAFETY ANALYSIS

This project involves the investigation of overall plant operational safety to assure the evaluation of this factor in the development of the final design.

9.0 Plant Safety Analysis

No work was performed under this subproject during the month of April 1958.

10.0 CRITICALITY EXPERIMENTS

Performance of criticality experiments on stainless steel clad UO_2 fuel elements at differing water-to-metal ratios are included in this project. Reactivity parameters and control rod effectiveness are to be determined.

10.0 Criticality Experiments

Temperature coefficient studies in the cylindrically loaded core were completed. Preliminary data analysis indicates the temperature coefficient is on the order of:

$$-1 \times 10^{-4} \Delta k/k \text{ -- } {}^{\circ}C \text{ in the range of } 60^{\circ} \text{ to } 70^{\circ}C$$

Flux measurements and reactivity changes were measured with the core containing stainless steel straps to simulate the fuel assembly makeup. A flux depression greater than 10% was observed in the region occupied by the stainless steel.

Cadmium ratios were established for indium, dysprosium and gold. No significant position dependence was found. The values obtained were: indium, 2.1; gold, 1.8; dysprosium, 23.

For 2.7% enriched uranium foil, the cadmium ratio was found to be 5.3 in the moderator and 4.7 in the fuel.

The reactor was loaded into a slab configuration for purposes of comparing a true slab with the nearly square configuration of the first loading to critical, and with the cylindrical configurations for other measurements. With control rods No. 7 and 9 in use, a loading 81 fuel rods long and 32 fuel rods wide (2524 fuel rods) was 0.09% $\Delta k/k$ supercritical. Buckling data was also obtained.

The void coefficient of reactivity was measured and found to be about 1% $\Delta k/k$ per percent void.

The Hazards Summary Report for the 2.23:1 Yankee Critical Experiments was completed.

11.0 RADIATION DAMAGE EXPERIMENTS

Design, construction and installation of a pressurized water loop for in-pile irradiation tests in the MTR and the performance of radiation damage experiments to demonstrate irradiation stability of Yankee core elements are involved.

11.1 Design and Fabrication of In-pile Test Loop

The loop component design, except for fuel sample, was completed and all drawings were released to the fabricator. The procurement of loop component equipment continued on schedule.

11.2 Performance of Radiation Damage Experiments

Post-irradiation diameter measurements of all the tubes in WCAP-1-1, -1-3, -1-5 and -2-1 capsules were made. There was no change in the diameter of the fuel tubes within the limits of accuracy of measurement. No center melting was observed in any of the pellets in the first four capsules examined.

12.0 LONG LIFE FUEL EXPERIMENTS

Work performed under this project will be directed toward the proof-testing of the prototype fuel assembly.

12.0 Long Life Fuel Experiment

No work was performed under this project during the month of April 1958.