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YAEC-79

YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

MONTHLY PROGRESS REPORT

MAY, 1958

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

JUNE 20, 1958

WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DEPARTMENT
PITTSBURGH, 30 P. O. BOX 355 PENNSYLVANIA

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YAEC-79

Yankee Atomic Electric Company

Research And Development Program

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MONTHLY PROGRESS REPORT

for the period

May 1st to 31st, 1958

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by

I. H. Coen
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

June 20, 1958

APPROVED: A E Voysey

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Project Manager

Westinghouse
ELECTRIC CORPORATION
ATOMIC POWER DEPARTMENT
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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 May, 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 May, 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, and Monthly Progress Report YAEC-70 describe the work accomplished from the beginning of the program, June 6, 1956 to April 30, 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

A study was conducted to determine whether the grinding sludge remaining from centerless grinding of CRX pellets to final diameter could be used for repelletization without chemical reprocessing.

Attempts were made to press UO_2 powder to the required green density to produce satisfactory pellets for MTR test specimens.

1.3.1 End Closure of Fuel Rods

A study was completed to determine the advisability of procuring an automatic welding head for end closures to increase weld quality and consistency.

1.3.2 Joining Fuel Bundles Into Assemblies

An experiment to determine the stretching characteristics of AISI 304 weldrawn stainless steel tubing has been completed.

The possibility of a damaging reaction occurring during brazing among UO_2, Ni-P braze and AISI 304 stainless steel cladding was investigated.

A 56 inch test bundle with Yankee tube spacings and ferrule design was thermal-cycled 28 times from room temperature to 600°F in pressurized, neutral, degassed water.

The vertical pit furnace has been checked out and found to operate properly, giving a 9 ft. uniform heat zone within $\pm$ 20°F.

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

Both fuel and cladding materials are being procured for the fabrication of the process water and in-pile loop samples.

A concerted effort is being made to have six additional process water samples at the MTR Site by August 1, 1958.

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 May 1958.

2.2 Core Steady State Analysis

A new method of calculating the resonance escape probability was devised. This method is expected to give improved agreement with experimental data.

The study of two region non-cycled cores was extended with two dimensional calculations.

2.3 Core Kinetic Analysis

A draft was prepared of a topical report describing the loss of coolant flow studies made on the Analog Computer.

2.4 Control Rod and Chemical Poison Analysis

Work continued on the control rod program study.

Studies of control rod worths in two region non-cycled cores were continued.

2.5 Critical Experiment - Design and Analysis

Work is continuing on the analysis of reflector savings and buckling measurements.

Preliminary calculations of migration area from incremental fuel rod and partial water height data yielded values 10 to 20% below the predicted values.

The use of the DARED code for reduction of foil counts to flux data on the IBM-704 computer neared completion.

2.6 Irradiation Experiment - Design and Analysis

Calculations were performed to determine the unperturbed neutron flux required in the MTR for 0.294 inch diameter pellets and a zero gap so that a heat flux of 600,000 Btu/ft^2-hr through clad thicknesses of 15 and 21 would be obtained.

2.7 Shielding Analysis

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

2.8 Startup Experiment Assistance

No work was performed under this subproject during the month of May, 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

Rohm and Haas XE-154 ion-exchange resin (Li^+, OH- mixed bed) was found to leak alkali while in contact with high purity demineralized water.

Boric acid capacity tests were completed for Nalcite SBR and Rohm and Haas XE-78 ion-exchange resins under "high pH" water conditions.

The number 10 dynamic autoclave was charged with 11.2 pounds of 11 mil s.s. wire, as a source for crud generation, and a performance test of Rohm and Haas XE-150 resin was begun in the autoclave's newly completed low-pressure purification system.

3.3 Corrosion of Materials of Construction

Test No. 6 was completed in PAR Loop A under the following average conditions:

Test Duration, hr. = 714
T = 600°F
P = 1828 psig
Flow rate = 249 GPM
Flow velocity = 38.8 feet per second
H_2, ml (STP)/kg solution = 27.1
B (as boric acid), ppm = 1561
Li (as lithium hydroxide), ppm = 0.98
pH = 6.27

The final draft of a topical report on "Qualitative Evaluation of Stainless Steel Tube Bundles Exposed in High Temperature Boronated Water" was completed.

Preparations are being made to initiate test PARA-7 which will be performed under simulated MTR test conditions (200 ppm B with 3 ppm Li).

Metallographic examination of stressed and sensitized AISI 304 stainless steel specimens exposed in Tests PARA-3, PARA-4, and PARA-5 (1600 ppm B, 1600 ppm B cycled to 3 ppm B, and 3 ppm B, respectively, with no pH adjustment) has been completed.

Recent developments regarding the corrosion of Ag-In-Cd alloy were investigated.

A corrosion rate was determined after 21 days from the corrosion testing in a static autoclave of Ag-In-Cd control rod material in 600°F water at 1500 psi pressure containing 1600 ppm of B and pH adjusted to 5.8 ± .2 with potassium hydroxide.

A corrosion rate was determined after 14 days exposure from the corrosion testing in a static autoclave of Ag-In-Cd control rod material in 600°F water at 1500 psi pressure containing 1600 ppm of B and pH adjusted to 5.8 ± .2 with lithium hydroxide.

3.4 Interactions Between Chemical Absorbers, Corrosion Products and Fission Products

System difficulties have delayed the starting of tests to determine the effects of Van de Graaff electron beam irradiation on crud deposition at both neutral and elevated pH.

3.5 Decontamination and Waste Disposal Studies

The relation of exposure time to change in surface finish during basic permanganate-citrate decontamination of AISI 304 stainless steel was determined.

A study was made of the relation of exposure time to weight loss in the decontamination of AISI 304 stainless steel using a 2% hydrofluoric acid -5% ammonium persulfate solution at 140°F.

3.6 Crud Inhibition, Suspension and Removal

Engineering is continuing 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 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

Consideration was given to the mechanical redesign of the fuel assembly based on the results obtained from a study of fuel assembly bowing.

4.2 Control Rod Drive Mechanism

Coordination of the design and fabrication of the prototype positive engagement type control rod drive mechanism was continued with the Westinghouse Atomic Equipment Department.

4.3 Design of Core Support Structure and Fuel Handling Tools

The fuel assembly, control rod, and guide tube handling sockets were redesigned to accommodate a recently designed universal handling tool. The preliminary design of the tool was completed and the final design was initiated.

The quarter size model of the baffle was completed and is ready for test.

4.4 Design For Critical Experiment and Irradiation Tests

A stainless steel "poison tank" for Critical Reactor Experiment tests was designed, fabricated and delivered to the Westinghouse Reactor Evaluation Center.

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

The maximum volume of the core that can be in a condition of local boiling was calculated for use in hot channel coolant flow studies.

The heat transfer and stress analysis study of bowing in the reactor fuel assembly was continued. Alternate mechanical designs are being developed to overcome possible mechanical interference due to excessive bowing of the fuel assembly.

The internal void pressure in the hot fuel rod at operating temperature was calculated.

5.2 Hydraulic Design

A study of a loss of coolant flow accident due to the failure of two pumps was completed. In an additional study, the momentary loss of power to the remaining two pumps was considered.

6.0 CONTROL ROD DEVELOPMENT

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

6.0 Control Rod Development

Preparation continued om Topical Report YAEC-58, Silver, Boron-Carbide Cermets (5, 10, 15 and 20-32 weight percent Boron-Carbide).

Topical Report YAEC-59, Development of a Method for Roll Cladding Silver, Boron-Carbide Cermets With AISI 304 Stainless Steel, was completed and issued.

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

A study was initiated on the IBM-704 digital computer to simulate pressurizer operation in the controlled plant. The necessary equations have been written, and programming of the problem is now in progress.

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

Topical Report YAEC-33, Study to Determine Economic Main Coolant System Parameters for the Yankee Project, was completed and issued.

8.11 Reactor Handling Tools and Plant Shielding Analysis

The design of the plate and barrel handling mechanism mock-up for testing in the Ⓦ AFD High Bay Building Deep Pit was completed.

Work was begun on a similar mock-up for the head gasket and seal ring fixture.

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 project during the month of May, 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

Experiments with borated water as the moderator were begun and are continuing.

Migration area studies were made with varying boron concentrations up to approximately 0.1 gram per liter. Data reduction is progressing.

A variety of flux profiles was taken to evaluate reflector savings. Gold foil work and fuel unit scanning techniques are currently in use. A neutron sensitive "semiconductor" was used to axially scan the operating reactor. Further analysis and correlation of data is required before the release of quantitative results.

Two high level (400 watt) runs were made to provide supplementary AREA DOSE RATE measurements. It was convenient to study the DISADVANTAGE FACTOR during these runs.

Preliminary results of the PARTIAL WATER HEIGHT experiments with an infinite dry reflector were obtained.

Three axial scans were made of banked control rods using copper wire techniques. These data are being studied in conjunction with control rod worth measurements.

Individual and group control rod worth measurements were made by varying the boron concentration in the water moderator.

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 motor control center and the annunciator panel were completed and are ready to be shipped to the Lummus Company.

Preparation of final design drawings and procurement and fabrication of In-pile test loop components by the Lummus Company are proceeding on schedule.

11.2 Performance of Radiation Damage Experiments

Following is the accumulated unperturbed nvt of the remaining WCAP-2 capsules at the end of Cycle 103, May 5, 1958.

Capsule	Accumulated Unperturbed nvt
2-2	3.5×10^{20}
2-4	3.9×10^{20}
2-5	3.6×10^{20}
2-7	3.6×10^{20}

Capsules WCAP-2-2, 2-4, 2-5 and 2-7 were transferred to a higher flux position.

Top priority has been given to the preparation for photomicrographs of the cracked ferrule brazes that were reported in assembly WCAP-21. The second group of capsules, WCAP-1-2, 1-4, 2-3 and 2-6, is being prepared for examination.

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 May, 1958.