YAEC-89

MASTER

YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

MONTHLY PROGRESS REPORT

JUNE, 1958

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

JULY 20, 1958

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

Yankee Atomic Electric Company

Research And Development Program

YAEC-89

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MONTHLY PROGRESS REPORT

for the period

June 1st to 30th, 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

July 20, 1958

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ABSTRACT

This report describes the work performed or coordinated by the Westinghouse Atomic Power Department for the Yankee Atomic Electric Company under Research and Development Subcontract No. 1 of USAEC-YAEC Contract AT(30-3)-222, during the month of June, 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 June, 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 Reports YAEC-70 and YAEC-79 describe the work accomplished from the beginning of the program, June 6, 1956 to May 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

Studies were made of the effect of green and fired density on the fired diameter of pellets.

1.3.1 End Closure of Fuel Rods

The installation of a new AIRCO automatic welding head has resulted in the production of end closure welds that are superior to those formerly obtained.

1.3.2 Joining Fuel Bundles into Assemblies

The first 36 tube test bundle, 52 inches long, having ferrule spacers coated with nickel phosphorus and Nicrobraz-50 slurry brazing compounds was brazed in the vertical pit furnace. After obtaining dimensions, the bundle was destructively tested for joint fracture loads. The next bundle will be brazed at a higher temperature and held for a longer time in order to obtain a comparison with present results.

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

Three prototype fuel capsules containing depleted UO_2 have been stretched. The three capsules are currently being evaluated for uniformity of stretching and will be used for a 30 inch prototype MTR loop bundle assembly. All the material for the MTR irradiation test specimens is available with the exception of the 10 and 27% enriched uranium dioxide.

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 June, 1958.

2.2 Core Steady State Analysis

The results of lifetime calculations of two enrichment cores are being compared with those from uniform cores.

CANDLE II, a revised form of the CANDLE program for analysis of reactor core burnup on the IBM-704 computer, was operationally analyzed. This program permits two group computation of reactor burnup as well as the four group computation previously available with CANDLE I.

2.3 Core Kinetic Analysis

An analog computer study of various procedures for simulation of heat transfer from the fuel pellet during power transients is in progress.

2.4 Control Rod and Chemical Poison Analysis

A preliminary study of the effect of a reference control rod program involving step wise outside-in rod withdrawal on flux distribution has been completed. A comparison study was begun of control rod effectiveness in two cores having different enrichments.

2.5 Critical Experiment - Design and Analysis

A calculation method which includes epithermal fissions has been devised which gives close agreement with the measured critical size of the CRX 3:1 water/uranium core. The method will be used in an attempt to predict the 2.23:1 water/uranium critical size.

A revision was made of the method used to calculate neutron age. Lower age values were obtained which were in closer agreement with experimental results.

Preparation of a topical report describing the JOFIT code was begun.

The draft of Supplement No. 2 to YAEC-31, Yankee Critical Experiment Hazards Summary Report, was prepared.

2.6 Irradiation Experiment - Design and Analysis

Calculations were performed to determine the optimum position of pellets to obtain a uniform heat flux in a 30 inch long capsule for MTR in-pile loop experiments.

A determination was made of the change in heat flux that would result from the change in diameter from 0.290" to 0.294" of 10 and 27% enriched pellets in the 30 inch long MTR in-pile test loop capsule.

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

Boric acid capacity tests were completed for Nalcite SBR and Rohm & Haas XE-78 resins under "high pH" water conditions. Water pH was adjusted to 10 with LiOH prior to boric acid addition. Flow through the columns was maintained at 25 ml/min (1.9 GPM/ft^3), and 130°F. At the conclusion of the test, the boric acid was eluted from the resin with Na OH. Analysis of the eluent was used to indicate the resin's total capacity. The experiment is being repeated at lower boric acid concentrations.

3.3 Corrosion of Materials of Construction

Test No. 6 in PAR Loop "A" was completed in which materials of construction were exposed for one month at 600°F. and 1800 psi to shutdown boric acid solution with sufficient lithium (less than 1 ppm) to raise the pH of unboronated water from 7.0 to 10.0.

Test No. 7 in PAR Loop "A" was initiated with water composition of the same type as that which will be used in the MTR chemical stability test (200 ppm B as boric acid and 3 ppm Li as lithium hydroxide). A one-month test is now in progress.

Dynamic autoclave tests are continuing comparing corrosion rates on silver-indium-cadmium control rod alloy in shutdown boric acid solution to which has been added sufficient lithium hydroxide and potassium hydroxide to raise the pH of neutral water from 7.0 to 10.0.

Components to build the variable velocity by-pass test loops for PAR Loop "A" were received and shipped to the fabricator.

Fabrication of the sample holders for the PAR Loop "A" by-pass test loops was begun.

3.4 Interactions Between Chemical Absorber, 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. A schedule of six 100-hour tests has been established, three under irradiation and three control runs similar in every way but without irradiation. At present, two control runs and one run under irradiation have been completed. A second run under irradiation is in progress. Results obtained thus far are being evaluated.

3.5 Decontamination and Waste Disposal

Preparation of the draft of a topical report entitled, "A Study of Decontamination Agents for Use in the Yankee Reactor" is nearing completion. This covers the work done during the first phase of the Yankee Decontamination Project leading up to the selection of the basic permanganate-citrate procedure as the method offering the most promise for successful solution of Yankee primary loop contamination problems.

In an attempt to gain experience with the application of the basic permanganate-citrate decontamination procedure in situations more closely approaching actual reactor application, a dynamic loop cleanup trial was conducted using the MED Loop "B" at East Pittsburgh. A detailed report on the results obtained is being prepared.

3.6 Crud Inhibition, Suspension, and Removal

Several alternate engineering designs have been considered, heat transfer calculations made, and evaluated for the crud deposition loop (simulated steam generator condition) to operate in a side stream off PAR Loop "A". Selection of final design and preparation of working drawings was begun.

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

Fabrication of a newly designed control rod coupling section has been completed with the exception of the Inconel "X" springs which have been ordered.

The mechanical effect of a temperature difference from one side to the other side of a fuel subassembly is being studied.

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

Testing and evaluation of a quarter size model of the core baffle was begun.

The need for enlarging the coolant flow holes was determined in the center of the upper and lower core support plates on the basis of the results obtained from hydraulic loop tests.

4.4 Design for Critical Experiment and Irradiation Tests

Modifications were made in the designs of the 2.23:1 lattice aluminum core plates, cruciform support plates, and lucite spacer plates. The modifications were made in the Westinghouse APD Model Shop under Project 10.0.

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 topical report draft, "Studies of Thermal Behavior Under Loss of Pump Power Transient Conditions", YAEC-72, was completed.

A general study was completed of the various size cores which can be contained by the reactor vessel.

5.2 Hydraulic Design

The effect of local boiling on pressure drop and flow distribution in the reactor core was investigated.

A model study was conducted to examine the pressure drop relationships in a typical fuel rod assembly.

6.0 CONTROL ROD DEVELOPMENT

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

6.0 Control Rod Development

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

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

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

Coordination and evaluation of the Chemistry Program was continued.

8.11 Reactor Handling Tools and Plant Shielding Analysis

The conceptual design of the handling fixture for the guide tube hold-down and support plates and the upper core support barrel was completed.

Work was continued on a head gasket and seal ring fixture mock-up for testing in the Westinghouse 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 project during the month of June, 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

The void coefficient of reactivity in the borated core was measured.

Flux profile measurements were continued to further evaluate reflector savings and buckling.

A series of experiments using boron in the moderator were performed with the lithium-stainless steel cruciform. Epithermal absorption characteristics of stainless steel are to be evaluated by this technique.

Experiments on the peripheral fuel unit worth continued.

Preparation of a draft of a topical report covering the 3:1 Yankee critical experiments was continued.

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

Piping and component drawings prepared by the loop fabricator (the Lummus Company) were reviewed and tentatively approved. The welding specifications were reviewed and returned to the fabricator with recommendations.

11.2 Performance of Radiation Damage Experiments

The following activity was performed on the second group of capsules removed from MTR, identified as WCAP-1-2, -1-4, -2,3, and -2-6:

1. Capsule disassembly
2. Removal of end cap flux monitors
3. Flux monitors identified and shipped to Nuclear Science and Engineering Company
4. Visual examination of bundles and ferrules

The following post-irradiation work was performed on WCAP-1-4, and WCAP-2-3 specimens:

1. Disassembly of bundles into individual tubes
2. Measure diameter and length of all tubes in bundle assemblies
3. Visual examination of all tubes

In addition, a metallographic examination of the cracked ferrule brazes on WCAP-2-1 was completed. Fine hair line cracks were seen at approximately 6X magnification. Brazes that were cracked through showed evidence of very poor diffusion bonding. Photomicrographs were taken of the cracked braze and of an apparently good brazed joint.

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 June, 1958.