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YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

MONTHLY PROGRESS REPORT

JULY, 1958

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

AUGUST 20, 1958

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

DISCLAIMER

Yankee Atomic Electric Company

Research And Development Program

YAEC-92

MONTHLY PROGRESS REPORT

for the period

July 1st to 31st, 1958

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by

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Large Plant Engineering

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

August 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 July, 1958. YAEC Development Program Report, YAEC-61, Rev. 1 outlines the Research and Development Program for the period from July 1 to December 31, 1958.

INTRODUCTION

This report describes the Research and Development performed during July, 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-61, Rev. 1, 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, YAEC-65, and YAEC-87 describe the work accomplished from the beginning of the program, June 6, 1956 to June 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 Development

The production of 270 depleted pellets for use in MTR testing required removal by centerless grinding of only 1.55% material to reduce pellets to desired diameter.

Sintered density control will be improved if the pellet density specification is revised upward. As steps toward this goal, two small scale experiments were begun:

1. UO_2 pellets were made with sintering aids.
2. An experiment was begun to determine whether the reversals of densification at higher sintering temperatures reported previously were anomalous or are natural to UO_2 powders.

1.3.1 End Closure of Fuel Rods

Considerable difficulty was encountered as a result of high air pressure build-up in the end plug-disc compartment. During welding, the pressure forced air out through the weld causing blowholes to form. Blowholes have been eliminated by an end plug pressing-repressing operation which limits the interference fit enough to eliminate the pressure build-up.

1.3.2 Joining Fuel Bundles Into Assemblies

A 36-tube test bundle, 52-inches long, was assembled and brazed to check the effect of three hours at 1875°F on joint strength. Eighty-four percent of the joints were ductile and 16% were non-diffusion bonded. In addition, electroless nickel thicknesses of 0.0005 and 0.001 inches were evaluated.

Tube stretch development work during the past month was limited to manufacture of MTR fuel tubes.

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

All Yankee irradiation specimens have been designed and drawings completed. Dummy specimens were fabricated and sent to the Lummus Company so that operational tests on the WCAP-4 loop would be conducted. Manufacture of the actual specimens has been started.

All the required enriched pellets and depleted pellets have been received for the fabrication of the in-pile and process water samples

During the stretch-forming operations, a large percentage of the 27% enriched pellets cracked. It was decided to reprocess the materials.

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 July, 1958.

2.2 Core Steady State Analysis

A detailed comparison of the equations involved in the CANDLE program for determining reactor lifetime with those in the CAP-1 program was made. An analysis of the effects of utilizing SOFOCATE thermal neutron cross sections in the CAP-1 fuel burnup equations was undertaken in order to evaluate the usefulness of a combined SOFOCATE - CAP-1 computer program.

A series of cursory design studies was made to evaluate the nuclear factors affecting bowing of the Yankee fuel assemblies.

2.3 Core Kinetic Analysis

The analog computer study of procedures for simulation of heat transfer from the fuel rod to the coolant in the Yankee core was completed.

The preparation of the final draft of topical report YAEC-83, Loss of Flow Accident by Computer Analysis in the Yankee Reactor, was completed.

Additional factors which are expected to affect the temperature coefficient of the Yankee core are being recalculated for incorporation in a YAEC topical report now being prepared.

2.4 Control Rod and Chemical Poison Analysis

A study of control rod programming in the Yankee core has been extended using improved techniques.

2.5 Critical Experiment-Design and Analysis

Two different types of control rod worth measurements for the 3:1 w/u ratio critical core were analyzed.

The JOFIT code is being subjected to final tests prior to the issuance of topical report YAEC-86, JOFIT - A Least Squares Bessel Jo Fitting Program for the IBM 204 Computer.

A study was completed of the variations in the transient behavior of the Yankee Critical Experiments produced by changing the values of various core parameters.

2.6 Irradiation Experiment - Design and Analysis

A study was made to determine the best arrangement of three different fuel enrichments along a 30 inch specimen in order to achieve a uniform heat flux.

Calculations were made to predict the variation of burnup as a function of the unperturbed flux.

2.7 Shielding Analysis

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

2.8 Startup Experiment Assistance

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

Boric acid capacity experiments were completed using a 40 ppm boron solution and Nalcite SBR and Rohm and Haas XE-78 anion resins. The XE-78 had a boric acid capacity of 3.9 lbs/ft^3 and the SBR had a capacity of 3.3 lbs/ft^3 at a flow rate of 1.9 gpm/ft^3 at 130°F with the pH adjusted to 10 with lithium hydroxide.

Decontamination factors were determined for Rohn and Haas resins XE-150 (H^+, OH^-), XE-154 (Li^+, OH^-) and XE-170 ($NH_4^+ \cdot OH^-$) using a 0.012 ppm cobalt solution at pH 8.5.

The results indicate that the tested resins have a greater affinity for cobalt than for either or both of the remaining corrosion constituents, iron and chromium.

Hydraulic characteristics of XE-150 and "Neva-Clog" filter medium, the respective reference resin and underdrain filter for the Yankee Reactor, were investigated.

3.3 Corrosion of Materials of Construction

Corrosion Test No. 7 in PAR Loop "A" was completed. This test was set up to simulate early MTR in-pile test loop operation.

Screening corrosion Test No. 8 in PAR Loop "A" is in progress under the following nominal conditions: T - 600°F; P = 1850 psig; Flow velocity - 38 feet per second; H_2, ml (STP) per kg = 25/35; B (as boric acid) ppm = 1600 - 5 (2 cycles); Li (as lithium hydroxide), ppm = 1; pH - 10 - 6 (2 cycles).

Corrosion tests of Ag-In-Cd control rod material in 600°F water (1525 psi pressure) containing 1 ppm of Li as LiOH were performed in a dynamic autoclave system. Corrosion tests of Ag-In-Cd control rod material, plus 2% tin, run under the same conditions with no boric acid indicated a marked inprovement in exposed weight change.

The pH control agent selection test has been terminated after 1300 hours of testing in static autoclaves.

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. In this project, a schedule of six - 100 hour tests has been established, three under irradiation and three control runs similar in every way but without irradiation. During the last month the scheduled irradiation and control runs have been completed. Re-run of one control run is scheduled for the coming month. Analysis of the data, and the completed runs are scheduled for completion within the coming month.

3.5 Decontamination and Waste Disposal

A draft of a report on the chemical decontamination of the MED B Loop at East Pittsburgh was completed and is in review.

Thirty, two foot lengths of contaminated stainless steel pipe from the original S1W steam generator was received for use in decontamination work at APD.

A series of experiments were begun in an attempt to find a complexing agent which would be stable in a basic oxidizing solution of sufficient potential to render Yankee type crud films soluble. The goal of this study is to develop a modified basic-permanganate-citrate procedure which would employ a single solution instead of the two reagents now suggested. This would greatly simplify the decontamination process with respect to waste disposal and solution handling.

3.6 Crud Transport and Deposition

Work continued on the engineering of the crud study side stream section to be added to PAR Loop "A". Selection of design from several alternate designs was made, and a flow diagram and a sketch of the crud collector section were prepared. Calculations were completed for the crud collector considering several dimensional alternates and flow schemes.

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

Inconel X springs were assembled in the control rod coupling section. Testing of the modified couplings indicated the design to be satisfactory.

The control rod handling socket is being remachined in order to adapt it to the new universal handling tool. This operation will complete the control rod for the Yankee deep pit test unit.

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

A stress and deflection test of the quarter size model of the Yankee Baffle was completed. A dimensional analysis study of the experimental data is now in progress. Additional tests are planned.

The final draft of YAEC-77, Deflection and Stress Analysis of the Yankee Core Structure, was revised to include comments by Yankee and Large Plant Engineering. The report was completed.

4.4 Design for Critical Experiment and Irradiation Tests

Six sets of cruciform fuel rod supports (0.405 inch spacing) were 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 temperature of the center of the fuel, the average temperature of the hottest pellet, and the average temperature of the fuel in the core were calculated for various thermal power outputs of the core.

Programming of the loss of flow accident on the IBM-704 computer was started and is now being checked.

5.2 Hydraulic Design

Pressure drop calculations of the flow and pressure relationships through a fuel rod assembly were investigated in light of experimental data obtained on a fuel rod assembly model.

Preliminary tests on the hydraulic dashpot indicated no major design changes are necessary.

The heat transfer coefficient in a flow cell distorted due to bowing of the fuel subassembly was calculated

6.0 CONTROL ROD DEVELOPMENT

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

6.0 Control Rod Development

Considerable investigations have taken place during the past month on the advisability of nickel plating the Ag-In-Cd control rods. Two processes are being investigated; namely,

1. Where the nickel is electrolytically deposited directly on the base metal, and

2. A plate of copper is first deposited onto the base metal and then a plate of one to two mils thickness is deposited onto the copper plate.

In the case of the copper nickel plate, the material can be metallurgically bonded by annealing. It is expected that the nickel plate will aid materially in overcoming the problem of corrosion in water that has dissolved oxygen present.

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

Programming of the digital computer study (PRESTO) of the Yankee pressurizer characteristics was essentially completed, and debugging of the resulting program is 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 comtemplated reactor.

8.5 Chemical Handling and Control Systems

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

8.11 Reactor Handling Tools

Work was begun on the redesign of a head gasket and seal ring handling fixture to provide latching and unlatching for all the gasket and seal ring mechanisms by means of single vertical motion.

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 July, 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 temperature coefficient of reactivity study on the borated core completed the experimental program on the 3:1 water/uranium metal volume ratio Yankee CRX core. The reduction of this data is still in process.

Installation and testing of the 2.23:1 core plates was completed. The core was loaded in eight steps to a critical size of 2981 fuel rods in a cylindrical configuration containing five control rod followers. The clean critical size has been deduced to 3047 fuel rods in cylindrical configuration with no control rod followers in the core.

Peripheral fuel rod worth in the new core has been studied.

Other experiments on the 2.23:1 core which have been completed but not yet evaluated include:

Void Coefficient

Temperature Coefficient

Radial and Axial Flux Plots

Uranium Foil Activation.

A topical report covering the 3:1 Yankee Critical Experiment core has been drafted and is now being reviewed prior to publication.

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

A progress check and conference was held at the fabricator's plant with representatives from MTR, Yankee Atomic Electric Company, Lummus Company, and Westinghouse APD.

11.2 Performance of Radiation Damage Experiments

During the month, capsule WCAP-2-4 was discharged from the MTR at the end of Cycle 108. This capsule has been sent to KAPL in one of their casks for post-irradiation examination.

At the end of Cycle 106, July 7, 1958, the remaining WCAP-2 capsules received the following accumulated unperturbed thermal nvt:

Capsule	Accumulated Unperturbed nvt
2-2	5.9×10^{20}
2-4	6.3×10^{20}
2-5	6.0×10^{20}
2-7	5.8×10^{20}

Knolls Atomic Power Laboratory was visited to examine the work being performed on WCAP samples 1-4, 1-2, 2-3, and 2-6. A summary of the examination on the above four capsules is as follows:

1. Fine, hairline cracks were observed in the ferrule brazes at 4 to 6X magnification that were undetected by the unaided eye.

2. Helium in appreciable proportions was identified in the gas sample after burn-ups of approximately 10,000 MWD/T.

3. A high percentage of water vapor was determined by analysis. It is postulated that the high percentage of water vapor comes from insufficient drying of the fuel pellets or protection after drying.

4. Xe and Kr release was a small proportion of the gas sample.

5. No grain growth was observed within a fuel pellet.

6. No significant fuel tube diameter or length changes were observed.

7. No bowing or bulging of the fuel tubes was observed.