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MASTER

YAEC-103

YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

# MONTHLY PROGRESS REPORT
## NOVEMBER, 1958

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

DECEMBER 20, 1958


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


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Yankee Atomic Electric Company YAEC-103

Research And Development Program

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MONTHLY PROGRESS REPORT

for the period

November 1 to 30, 1958

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by

R. W. Garbe
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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

December 20, 1958

APPROVED: [signature]

A. E. Voysey
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 the Yankee Atomic Electric Company under Research and Development Subcontract No. 1 of USAEC-YAEC Contract AT(30-3)-222, during the month of November, 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 the month of November, 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, YAEC-87 and YAEC-97 and Monthly Progress Report YAEC-101 describe the work accomplished from the beginning of the program, June 6, 1956 to October 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 Development

#### $UO_2$ Powder and Pellet Specifications

$UO_2$ powder and pellet specifications were revised and issued.

#### L/D Ratio Studies

Consideration was given to the optimum L/D ratio for Yankee first core pellets. Although ground 0.6 inch long pellets were found to be less expensive than unground 0.3 inch long pellets, a study of the factors which presently influence the cost of manufacture of pellets longer than 0.6 inches resulted in the conclusion to maintain the Yankee first core pellet length at 0.6 inches.

#### $UO_2$ Powder Density

Confirming results previously reported, all indications are that pellets of high green density are much more likely to laminate when sintered to a high temperature or for a long time than pellets of lower green density. However, the temperature, the time, and the sintered density at which this reversal ("solarization") of density occurs varies greatly from powder to powder.

Since density becomes asympototic to some value below 100% of theoretical density, excellent density and dimensional control can be obtained by sintering a sufficient length of time or at a sufficiently high temperature to assure reaching the minimum specified density.

#### Hour-glassing

New data obtained during the reporting period confirm that hour-glassing may be minimized through choice of powder, pressing pressure, and lubricant.

#### Fabrication of Pellets for Fuel Assembly Prototype

Fabrication of pellets for use in the fuel assembly prototype will be performed on a trial run basis to assure that a $UO_2$ powder has been chosen which will be uniform from lot-to-lot, that the most economical powder preparation procedure will be employed, that all unnecessary operations will be omitted, that dimensional control will minimize grinding losses, and that losses will be low.

1.3.1 End Closure of Fuel Rods

Preparation of the first draft of a topical report on fuel tube end closures was completed.

1.3.2 Joining Fuel Bundles Into Assemblies

A detailed dimensional survey was made of an experimental disc-compartmented, 36-tube subassembly 60 inches long filled with $Al_2O_3$ pellets and brazed for one hour at 1850°F. The tube spacer ferrules were coated with electroless nickel phosphorous brazing alloy. After autoclaving and dimensioning, the subassembly was sectioned and pulled apart for brazed joint fracture load data.

On the next subassembly, small tubes and tube segments will be brazed to the corner tubes to provide additional corner tube strength.

The first draft of a topical report "Fabrication of YAEC Fuel Elements" was completed.

1.4 Fabrication of Fuel Elements for Critical Assembly and Irradiation Tests

In-Pile Loop Specimens

The six irradiation test specimens were shipped to the Lummus Company for insertion in the in-pile and out-of-pile sections of the WCAP-4 in-pile test loop.

Control rod samples plated with nickel, both defected and non-defected, were fabricated and sent to the Lummus Company with the fuel tube specimens.

A topical report discussing the fabrication of these test specimens is being prepared.

The various procedures and manuals for the MTR loop experiments with the exception of the hazards manual and operating manuals have been completed and sent to the MTR for approval.

## 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 November, 1958.

### 2.2 Core Steady State Analysis

The control rod program to be selected for the Yankee reactor will probably involve relatively rapid withdrawal and insertion of rods in regions in which some rods are fully inserted and others are partially withdrawn. When a rod is removed the absence of Xe-135 in the immediately adjacent area has been calculated to cause a 2% increase in the flux peaking hot channel factor.

A study is being made to examine the power flattening due to nonuniformity of Xe poisoning, water temperature and Doppler broadening.

A boundary search CANDLE code is being run to study the lifetime behavior of the axial hot channel factor in a core with blanket control.

### 2.3 Core Kinetic Analysis

The draft was prepared of a topical report describing the analog computer study of startup transients in the Yankee reactor.

A study was made of the digital computer code which has been developed for calculating flow redistribution due to localized boiling in the reactor core. Several changes will be made before application of this program to the study of partial loss of flow in the Yankee Reactor.

### 2.4 Control Rod and Chemical Poison Analysis

A calculation of the fast neutron absorption expected in a boron 10-stainless steel control rod was performed.

The flux synthesis technique of combining one dimensional flux distributions, used in a previously completed control rod programming study, was verified by comparison with PDQ three dimensional radial-axial results.

Rod worths and reactivities, as functions of enrichment, were calculated for the Yankee uniform first core using procedures verified by Westinghouse Reactor Evaluation Center Critical Reactor Experiment results. The results appear to indicate that there is adequate control with 24 control rods and that to obtain the desired lifetime of 10,000 hours a fuel enrichment of at least 3.2% is needed.

2.5 Critical Experiment - Planning and Analysis

Results were obtained from experiments designed to show the reactivity effect of resonance neutron absorptions in stainless steel, performed by comparing the worths of a steel rod and a Li-Al rod.

Fast neutron absorption cross sections were obtained from control rod worth analyses.

Values were obtained for the three Yankee CRX cores using the simple four factor analysis from an analysis of migration area experimental results obtained by peripheral fuel rod worths and core poisoning. Analysis of the same data using the split group model gave comparable values.

2.6 Irradiation Experiment - Design and Analysis

Extensive contributions were provided for the nuclear sections of descriptive and hazards reports for the WCAP-4 In-Pile Test Loop to be installed in the MTR.

2.7 Shielding Analysis

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

2.8 Startup Experiment Assistance

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

Preparation was completed of the draft of a topical report entitled, "Testing of Ion Exchange Resins for Use in the Yankee Reactor."

The resin evaluation program was continued with the testing of Rohm and Haas XE-154 mixed bed resin in the dynamic autoclave purification system. In the current experiment no filters or other aids to resin purification are being employed.

In an effort to determine the effect the presence of dissolved boric acid exerts on the ability of XE-154 resin to control and stabilize pH of coolant circulating through an XE-154 resin bed, a test simulating Yankee startup conditions was performed. The test was initiated with water containing 1600 ppm boron as boric acid and less than 1 ppm lithium. The procedure included 3 system volumes of bleed and feed, followed by 3 system volumes of circulation through an anion resin (XE-78), with subsequent passage through a Li-OH mixed bed resin (XE-154) until column inlet and outlet pH was equal. The pH stabilized at 10.4.

A second test duplicating the first, but using water adjusted to pH 10 with lithium for bleed and feed instead of demineralized water, showed a final pH of 10.6.

### 3.3 Corrosion of Materials of Construction

Autoclave Corrosion Rate Studies Under Simulated Reactor Startup and Shutdown - Low Oxygen

One dynamic autoclave test to determine the effect of concentration cycling of boric acid on corrosion of Yankee primary system materials is continuing. Nominal test conditions are:

| | |
|---|---|
| Flow velocity | - 0.1 fps |
| Temperature | - 600°F |
| Pressure | - 1525 psig |
| Water Chemistry | - pH - 5.8 - 10 (controlled with 4 ppm LiOH) |
| | B - 1600 ppm to 5 ppm cycled up and down every 14 days |
| | $O_2$ - 0.05 ppm |
| | $H_2$ - 30 cc/Kg (STP) |

The total time under test has now reached 24 days. Results will be reported at the end of the 56 days total test time.

Dynamic Autoclave Testing of Silver-Indium-Cadmium

The first month of dynamic autoclave testing of silver-indium-cadmium was completed. Specimen coupons of bare metal, nickel plated, and defected nickel plated control rod alloy were exposed to lithiated 600°F water, in the presence of hydrogen and low oxygen. Samples of the coupons were removed and sectioned.

Reaction of Soluble Oxygen with Hydrazine in Borated Water

The following experiments were conducted to determine the effect of hydrazine as an agent for removal of dissolved oxygen in water containing 1600 ppm boron as boric acid:

1) Effect of Temperature on Hydrazine - Oxygen Reaction

   A reaction was effected between water containing 1600 ppm boron as boric acid and 7 ppm of dissolved oxygen and an equimolar amount (7 ppm) of hydrazine in a dynamic autoclave.

2) Effect of Temperature on Dissolved Oxygen Content in Absence of Hydrazine

   Water containing 1600 ppm boron as boric acid and 6.1 ppm of dissolved oxygen was heated in a control rod drive mechanism housing with no hydrazine added.

3) Effect of Time at 200°F on Dissolved Oxygen Content in the Presence of Hydrazine

   Water containing 1600 ppm boron as boric acid was maintained at 200°F in a glass vessel. In the presence of 21 ppm hydrzine, the oxygen content in 10 minutes dropped from 7.1 ppm to 0.6 ppm.

   This test is to be repeated at 250°F.

Westinghouse APD Loop "A" Tests

Westinghouse APD Loop "A", Test No. 10, involving corrosion rate studies at elevated pH, is in progress. This test is scheduled for shutdown the week of December 15. A purification system will be installed during the shutdown period.

3.5 Decontamination and Waste Disposal

The first draft of a topical report entitled, "Development of Agents and Procedures for Decontamination of the Yankee Reactor Primary Coolant System" was completed.

A study has been undertaken of the problems associated with the disposal of the decontamination reagents after use in reactor primary coolant system cleanup. It is anticipated that this study will result in a series of recommendations concerning the ultimate disposal of the comparatively large quantities of waste cleanup compounds which will be produced in conjunction with the actual plant decontamination operation.

Some difficulty has been encountered in dissolving the recommended quantity (3 weight percent) of potassium permanganate crystals during makeup of the basic permanganate decontamination reagent. In an effort to avoid possible clogging of decontamination system lines with undissolved crystalline potassium permanganate, an evaluation was made of the advisability of using $KMnO_4$ powder in place of the more common crystalline form of this compound.

## 3.6 Crud Transport and Deposition

Work continued on an experiment to determine the effect of heat flux on crud deposition. The heat exchanger used by the Westinghouse Radiation and Nucleonics Laboratory for their crud studies was sectioned. Attempts to descale one of these sections by means of the alkaline permanganate solution proved ineffective. Because of the limitations of the alkaline permanganate solution, the descaling procedure has been changed.

## 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 final draft of a report on the thermal deflection of a fuel assembly as caused by linear and non-linear temperature gradients is being written. An analytical method which predicts thermal deflections to within 5% of experimental results was developed.

The handling sockets at both ends of the fuel assembly were redesigned for increased strength and simplicity.

### 4.2 Control Rod Drive Mechanisms

The test stand for the Latch Type Magnetic Jack Control Rod Drive Mechanism was received and installed on schedule in the Westinghouse APD High Bay Building. Tests were made to determine pulley friction and load capacity.

All components for the mechanism were received from the Westinghouse Atomic Equipment Department. Critical dimensions of all moving parts were measured and recorded for future reference; the mechanism was prepared for operation.

The power supply for the mechanism was received from the vendor. Initial operation of the mechanism coil stack, connected to the power supply, was successful.

An automatic cycling device is now being fabricated in the Westinghouse APD Model Shop. This unit will be used to cycle the operations of the mechanism.

### 4.3 Design of Core Support Structure and Fuel Handling Tools

Work continued on the fabrication of the Yankee Universal Handling Tool in the Westinghouse APD Model Shop. A modification was made to the tool handling head to improve the self-centering action.

### 4.4 Design for Critical Experiment and Irradiation Tests

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

The parameters affecting the thermal design of the Yankee first core fuel rod have been assembled in the first draft of a topical report "Significance of Pellet-Clad Clearance in the Thermal Design of the Yankee First Core."

### 5.2 Hydraulic Design

A ferrule-type fuel assembly model was fabricated and tested in the 300 gpm hydraulic test loop. A section having ferrules at every space and one with alternate ferrules missing were studied. Each ferrule array indicated a substantially lower pressure drop than the strap array. The final draft of topical report YAEC-75 "Model Study of the Pressure Drop Relationships in a Typical Fuel Rod Assembly" is being revised to incorporate the ferrule studies as well as a vibration study of the fuel rods.

Work was completed on the Yankee control rod shock-absorber. Several variable orifice type dashpots were designed, fabricated, and tested. Test data revealed that approximation of the theoretical contour by straight line tapers was justified. Based on test data, a triple taper hydraulic shock absorber was desigend for the Yankee reactor control rods. The first draft of a topical report on the dashpot studies was prepared.

## 6.0 CONTROL ROD DEVELOPMENT

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

### 6.0 Control Rod Development

Ag-In-Cd material was procured from KAPL and will be used to fabricate a prototype Yankee control rod.

Tensile and stress-rupture specimens of Ag-In-Cd were tested at 550°F. In these tests, a comparison was made between annealed specimens and annealed and corrosion-tested specimens. A comparison of the tensile results showed no harmful effects from limited internal oxidation.

Bend specimens of Ag-In-Cd were electroplated with 1) 1-1/2 mils of nickel, 2) copper strike plus 1-1/2 mils of nickel and 3) 1/2 mil of copper plus 1-1/2 mils of nickel.

An order was placed for an alloy of 80.5% silver, 15% indium, 4% cadmium and 0.5% tin. Tin has been added in an attempt to increase the corrosion resistance of Ag-In-Cd to environments containing oxygen as a continuation of preliminary investigations performed by the Bettis Plant personnel.

## 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 November, 1958.

## 8.0 PLANT 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 November, 1958.

8.11 Reactor Handling Tools

A portion of the material for the plate and barrel handling mechanism mockup was ordered and minor changes were made in the design of the mockup. Fabrication of the mockup was begun by the Westinghouse APD Model Shop.

## 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 November, 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

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

## 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 Lummus Company has completed all the items that needed correction and action except for those items that need checking out at loop temperatures. Failure of the temperature control valve in the WCAP-4 in-pile loop during startup of a final test at the Lummus Company prevented completion of the test. Failure was due to leakage through a bellows seal in the three-way control valve.

WCAP-4 in-pile loop test analytical procedures proposed by Westinghouse APD were discussed with personnel at the Chemical Processing Plant site in Idaho and were found to be satisfactory to them. Periodic check samples will be submitted for Westinghouse APD duplication during the in-pile test.

The mechanical design sections of the In-Pile Test Loop Operating and Hazards Manual were completed.

### 11.2 Performance of Radiation Damage Experiments

Process water capsule WCAP-2-13 was inserted into MTR position A-29-NE during Cycle 113 shutdown.

Process water capsules WCAP-2-8 and 2-11 are scheduled for insertion into ETR position AJ-18 and WCAP-2-9 and 2-12 are scheduled for insertion into position AI-18 at the Cycle 10 shutdown. These two positions have an average flux of approximately $5.0 \times 10^{13}$ nv, with a peak in the range of $6.0 - 6.5 \times 10^{13}$ nv.

Preparation of a report of the results of the post-irradiation examination of process water capsules WCAP-1-2, 1-4, 2-3 and 2-6 is in progress by KAPL personnel. The examination of capsules WCAP-1-1, 1-3, 1-5 and 2-1 was completed by KAPL and a report covering the results is being prepared.