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YAEC-99

YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

MONTHLY PROGRESS REPORT
SEPTEMBER, 1958

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

OCTOBER 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-99

MONTHLY PROGRESS REPORT

for the period

September 1 to 30, 1958

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

October 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 September, 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 September, 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 (UO₂) 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 and Monthly Progress Reports YAEC-92 and YAEC-95 describe the work accomplished from the beginning of the program, June 6, 1956 to August 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

A study was begun to determine the effects of varying the length-to-diameter ratio of pellets, forming pressure and time, and additives on the "hour-glassing" of pellets.

Considerable data have been developed from sintering studies of various samples of UO_2 such as the occurrence of unsintered cores in pellets sintered for two hours at 1425°C and variations in surface tone which will be correlated with X-ray patterns and corrosion resistance values.

1.3.1 End Closure of Fuel Rods

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

1.3.2 Joining Fuel Bundles Into Assemblies

A 95 inch experimental Yankee sub-assembly was successfully brazed at $1890 \pm 10^\circ\text{F}$ for three hours then furnace cooled. All of the joints, outside ferrules and control rod rubbing strips showed a brazed fillet.

Dimensional surveys and joint strength studies indicate that two brazing techniques appear feasible for sub-assemblies. One technique is chemical plating Ni-P on the ferrules backed up with Microbraz 50 slurry and the other technique is simply plating the ferrules with Ni-P.

Experimental loading and stretching have been continued using full length Yankee tubes. It was found that a vacuum drawing of the pellets into the tube was required since a column of pellets longer than 5 to 6 feet produced excessive friction. Pellet chipping and dusting of pellets which increase loading friction were also minimized by vacuum loading. Planning has been directed toward incorporating the loading operation in the same machine that will stretch the tubes.

1.4 Fabrication of Fuel Elements for Critical Assembly and Irradiation Tests

All of the in-pile test loop and process water samples were completed except for the in-pile loop specimen containing 27% enriched pellets.

The 27% enriched fuel specimen has been fabricated and will be brazed into a fuel bundle.

Several attempts were made to electro-nickel plate silver-indium-cadmium specimens for the in-pile loop test. A satisfactory plate has not been obtained to date. Additional samples are being nickel plated by a vendor and it is expected that the plated Ag-In-Cd specimens will be available in time to ship with the loop to MTR.

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 September, 1958.

2.2 Core Steady State Analysis

Development of the Fortran code for recalculating the Dancoff correction used in resonance escape probability calculations has been completed.

Considerable progress has been made, using various computational procedures, in the development of information relating to values obtained for thermal neutron flux peaking in water slots. Fuel lattice - water channel configurations for which experimental data are available are being analyzed with both transport theory and diffusion theory in two and four groups. Macroscopic cross sections calculated by the MUFT IV multi-group code and the SOFOCATE code for averaging over a thermal neutron spectrum are being used.

2.3 Core Kinetic Analysis

The effect of xenon concentration redistribution on the readings of top and bottom core detectors was determined by the application of the PDQ computer code in radial and axial flux analyses.

A revised model for the analog computer simulation of reactor startup transients is being applied to a study of startup transients in the Yankee first core.

2.4 Control Rod and Chemical Poison Analysis

A control rod programming study for two region cores is now in progress.

2.5 Critical Experiment - Design and Analysis

Experimental power plots with control rod followers in place are being compared with both one and two dimensional calculated plots in an effort to evaluate the methods used in studying control rod programming.

A series of criticality calculations for the three water-to-metal volume ratios of the Yankee critical experiments has been concluded with the calculation of the critical mass of 4:1 water-to-uranium metal volume ratio core. The series of calculations was based on the inclusion of heretofore neglected effects such as epithermal fissions in U-235, and resonance capture of neutrons in the stainless steel clad. An analysis

of the results of these calculations in light of the experimental data is in progress; this analysis has led to the adoption of a somewhat modified expression for the resonance escape probability for U-238.

The experimental results for disadvantage factors obtained at the Westinghouse Reactor Evaluation Center are being compared with results obtained in diffusion theory and transport theory calculations for a unit cell of the fuel-water lattice. Application of the SNG computer program to a calculation of disadvantage factors using transport theory is in progress.

2.6 Irradiation Experiment - Design and Analysis

Calculations were performed to determine the MWD/T burnup in terms of unperturbed nvt for 2.7 and 5.4% enriched process water test specimens.

2.7 Shielding Analysis

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

2.8 Startup Experiment Assistance

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

Temperature stability experiments were conducted on Rohm and Haas XE-150 resin by means of water baths at the purification system operating temperature of 140°F and at an elevated temperature of 170°F. The higher-than-normal operating temperature was used to determine whether a temporary temperature excursion would be detrimental to the resin. Resin capacity was obtained by passing a sodium chloride solution through the resin column and monitoring the effluent with a conductance bridge in order to determine relative breakthrough values.

3.3 Corrosion of Materials of Construction

Screening Corrosion Test No. 9 in PAR Loop "A"

Test No. 9 in PAR Loop "A" involving the corrosion testing of materials of construction was completed under the following average conditions:

600°F temperature; 1834 psig pressure; 38.3 fps flow rate; 29.5 ml(STP)/kg of hydrogen; 2.5 ppm lithium; 12.4 ppm boron and a pH of 8.1.

Specimen data obtained from the test are being analyzed.

Simulated Reactor Startup and Shutdown Tests

Two dynamic autoclave corrosion tests are being performed to determine the effect of concentration cycling of boric acid on corrosion of Yankee primary system materials. One autoclave contains 1 ppm of Li as LiOH and the other contains no lithium.

Test conditions are: 600°F temperature, 1525 psig pressure, 0.1 fps flow rate; water chemistry - Test 1, pH - 5.8 to 10 (controlled with a constant concentration of 3.4 ppm LiOH); Test 2, pH - 4.8 to 7; 30 cc H₂/Kg to H₂O (STP); boron - 1600 ppm to 5 ppm cycled up and down every 14 days.

Results of 14 days testing indicate:

1. A slight decrease in corrosion rates of AISI 410 stainless steel and Inconel due to the presence of LiOH.

2. A slight increase in the corrosion rate of Ag-In-Cd alloy due to the presence of LiOH.
3. A slight increase in corrosion rates of AISI 410 stainless steel and Ag-In-Cd alloy due to the boric acid concentration cycling.
4. All other Yankee primary system materials were unaffected by pH cycling or the presence of LiOH.

Reaction of Soluble Oxygen with Hydrazine in Borated Water

Three tests were performed in dynamic autoclaves involving the addition of hydrazine solution to water containing 9145 ppm H_3BO_3 (1600 ppm B) and about 6.5 ppm O_2 at room temperature, mixing for 30 minutes, and then heating the autoclave. Cooled samples were taken at specific time intervals and the temperature noted. These were then analyzed for oxygen and hydrazine.

3.4 Interactions Between Chemical Absorber, Corrosion Products, and Fission Products

This project has supported a study of the effects of Van de Graaff electron beam irradiation on crud deposition, at both neutral and elevated pH. The study was completed and data were analyzed.

3.5 Decontamination and Waste Disposal

The decontamination study making use of contaminated SLW stainless steel steam generator tubes was continued.

Information was obtained on the following subjects:

1. Decontamination factor values
2. Effect of temperature on decontaminating ability
3. Effect of total exposure time on decontamination ability
4. Relationship between exposure time and decontamination factor values
5. Base metal attack rates
6. Effect of incomplete rinsing
7. Effect of dilution of the reagent
8. Variations in caustic concentrations

3.6 Crud Transport and Deposition

Calculations were made to determine the feasibility of running crud deposition experiments in PAR Loop "A" under simulated Yankee steam generator conditions.

It was decided not to assemble and install the equipment considered for this experiment on Loop "A", but rather to find out whether

preferential crud deposition exists at all on cooled surfaces. This will be accomplished by sectioning portions of the Van de Graaff crud deposition loop from (a) the heat exchanger and (b) thermally insulated piping immediately adjacent to the heat exchanger, and exposing the sections to the basic permanganate-citrate decontamination solution to determine whether any significant difference in total crud and corrosion product exists between the two situations.

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

Preparation of layout drawings of the compromise fuel assembly design is continuing.

A laboratory test setup was initiated for determining bowing and deflection of fuel subassembly caused by linear and non-linear temperature gradients.

An end nozzle arrangement of new design is being used with a new method of attachment to the fuel assembly. The new nozzle will permit handling of the fuel assembly by the new Universal Handling Tool.

The control rod handling socket was modified so it can be handled by the Universal Handling Tool. The prototype control rod is now complete and ready for use in the Yankee channel mock-up test program.

4.2 Control Rod Drive Mechanisms

The contract for the design and construction of a test stand for the control rod drive mechanism prototype is being negotiated with the Hamill Manufacturing Company.

The contract for building the power supply for the prototype mechanism was awarded to the Westinghouse Director Systems Department.

4.3 Design of Core Support Structure and Fuel Handling Tools

Design modifications on the quarter size model of the baffle were completed, after which it was again tested for deflection. Calculations are now being performed to determine the magnitude of deflections in the full size baffle, based upon the deflection measurements.

The Westinghouse APD Model Shop started fabrication of the hydraulically actuated Universal Handling Tool.

4.4 Design for Critical Experiment and Irradiation Tests

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

Values of the neutron fluxes to which the WCAP-2 samples were exposed were received from the MTR. Calculations are now in progress to determine the exposure conditions.

Calculations were begun to determine the axial position which will have the highest maximum-to-average power ratio in the hot channel during the core lifetime.

A hazards analysis was made on loss of flow accidents in the WCAP-4 In-Pile Test Loop. A report is being prepared.

5.2 Hydraulic Design

Hi-speed motion pictures were taken of the control rod hydraulic dashpot prototype to determine the effects of substituting a heavier spring. The heavier spring eliminated the precompression of the dashpot due to hydraulic forces, but the taper of the piston must be changed in order to attain optimum deceleration rates.

6.0 CONTROL ROD DEVELOPMENT

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

6.0 Control Rod Development

No work was performed under the project during the month of September, 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

The digital computer program for analysis of transients in the Yankee pressurizer has been completed and is being tested. This program will provide information required to develop an analog computer model for the pressurizer to be used in an analysis of the controlled Yankee Plant.

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 September, 1958.

8.11 Reactor Handling Tools

Redesign of the head gasket and seal ring fixture was completed.

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 September, 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

2.23:1 Water/Uranium Metal Core - Clean

Two runs were made to provide flux profiles of the clean core. Values for reflector savings and buckling were determined.

A flux profile was measured across a water slot. The peak-to-unperturbed flux ratio was determined.

The ratio of resonance capture to thermal capture in the U-238 was obtained.

The cadmium ratio of U-235 in the moderator and fuel was determined.

2.23:1 Water/Uranium Metal Core - Borated

Temperature coefficient measurements were made, and the data analysis is now in process.

Two runs were made to obtain flux profiles for reflector savings and buckling evaluations.

The disadvantage factor was evaluated with a boron concentration of 400 ppm in the water moderator. The core contained 4613 fuel rods while these measurements were made.

The peripheral fuel rod worth was established in a core containing 4745 fuel rods and a boron concentration of 400 ppm.

4:1 Water/Uranium Metal Core - Clean

Loading the core to critical and core adjustment to ascertain the just-critical core required five steps. It was determined to be 1300 fuel rods with no control rod followers at 18°C.

Flux profiles were established with the initial loading configuration (off center loading centered between control rods 1, 4, 6 and 5). These data are now being processed.

An experiment to evaluate resonance escape and thermal utilization was performed, and the data are now being processed.

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 test loop was tested for approximately 120 hours in the vendor's shop, after which it was found to be not acceptable because of several faulty components.

A discussion was held at the Lummus Company concerning the acceptance of the WCAP-4 loop for delivery during which a list of 54 items that needed correction and/or action was presented to Lummus personnel. Lummus, with Westinghouse APD assistance, is now working on correcting these items.

11.2 Performance of Radiation Damage Experiments

Proposed irradiation examination of Capsule WCAP-2-4 has been delayed at KAPL due to the renovations being made in the hot cell facilities. It is expected that KAPL will be able to examine the specimens starting November 1, 1958.