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YAEC-95

YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

MONTHLY PROGRESS REPORT AUGUST, 1958

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

SEPTEMBER 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-95

MONTHLY PROGRESS REPORT

for the period

August 1 to 31, 1958

by

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

September 20, 1958

APPROVED:

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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 August, 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 August, 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 and Monthly Progress Report YAEC-92 describe the work accomplished from the beginning of the program, June 6, 1956 to July 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

The draft of topical report YAEC-84, "Progress in the Development of a Process for Producing UO_2 Pellets", was completed.

Preliminary tests were performed to improve the sinterability of UO_2 powder, using additives and/or steam sintering, to obtain higher densities at lower sintering temperatures.

1.3.1 End Closure of Fuel Rods

X-ray studies of experimental end plug welds were performed to determine the amount of tungsten pick-up from the tungsten electrode in the welding head.

End plug welding procedures were rewritten incorporating the use of the new AIRCO automatic welding head.

End plugs were successfully welded to AISI 304 stainless steel, 15 mil tubing for irradiation specimens using the AIRCO welding head.

1.3.2 Joining Fuel Bundles Into Assemblies

Difficulties encountered in loading Al_2O_3 pellets into 92" long tubes delayed the start of stretching experiments on full-length fuel rods. A new loading technique using a vacuum system to draw the pellets and discs into the tube is being investigated and appears promising. In the initial loading studies, tubes that could not be hand loaded were readily loaded with the aid of the vacuum technique.

The feasibility of brazing end plugs into fuel tubes using a method similar to that employed for brazing the discs in compartmented fuel rods is being investigated.

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

Considerable distortion was encountered during the brazing of prototype MTR 24-inch, three-tube bundles consisting of rods in which the tubing was stretched onto the pellets.

To date, a total of 48 fuel rods have been stretched for use in the MTR irradiation program.

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 August, 1958.

2.2 Core Steady State Analysis

A detailed study was performed of the power peak about a control rod follower. A study was then initiated to compare the values of flux peaking adjacent to control rod followers with the values of flux peaking at the core-reflector interface as calculated by various procedures.

Comparison studies of various two-region uncycled and one-region uniform cores are continuing. The results of these studies will be included in a topical report.

2.3 Core Kinetic Analysis

The draft of YAEC-73, "Kinetic Parameters of the First Yankee Core: Temperature and Void Coefficients, Neutron Lifetime and Delayed Neutron Fraction", was completed. The report describes the calculation of these parameters and their effect on the temperature coefficient of reactivity.

2.4 Control Rod and Chemical Poison Analysis

The preliminary study of control rod programming in the Yankee first core was extended and checked by means of the PDQ code in two-dimensional geometry.

2.5 Critical Experiment - Design and Analysis

YAEC-86, a topical report describing the JOFIT code which calculates the radial buckling from flux measurement data, is being prepared in final form for publication.

Initial reflector savings measurements have been made in the Yankee Critical Reactor Experiment 2.23:1, water to uranium metal ratio core.

The results of an experiment comparing the worths of a stainless steel and a lithium-aluminum control rod in the CRX 3:1 W/U ratio core have been analyzed.

A power distribution plot in the CRX 2.23:1 W/U ratio core, obtained by fuel rod counts, was taken through two control rod followers.

2.6 Irradiation Experiment - Design and Analysis

It was determined that corrosion of the control rod samples which are to be placed in the MTR test loop would constitute only a negligible hazard from radioactive deposition.

2.7 Shielding Analysis

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

2.8 Startup Experiment Assistance

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

Decontamination factors were determined for Rohm & Haas resins XE-150 (H^+ , OH^-), XE-170 (NH_4^+ , OH^-), and XE-154 (Li^+ , OH^-) using 0.06 ppm iron solution and 0.007 ppm chromium solution. The pH was adjusted to 8.5 with NH_4OH for the XE-150 and XE-170 resins and with $LiOH$ for the XE-154 resin.

Decontamination factors were determined for Rohm & Haas resin XE-154 using a mixed corrosion product solution (concentration in ppm: Fe, 0.06; Cr, 0.007; Co, 0.012). pH was adjusted to 8.5 with $LiOH$.

3.3 Corrosion of Materials of Construction

Data were obtained from screening corrosion Test No. 7 in PAR Loop "A". This test was designed to screen Yankee materials of construction under conditions of cycled boron concentration in the presence of lithium hydroxide.

During the last period of operation, the loop heaters were not able to maintain the specified temperature. This is probably due to decomposition of the thermally conductive cement (Thermon) used to improve contact between the external strip heaters and the loop wall.

Corrosion tests of Ag-In-Cd control rod material in 600°F water (1525 psi pressure containing 1 ppm of Li as $LiOH$ with 30 cc $H_2/L H_2O$ (STP)) were continued in a dynamic autoclave system with a flow of 0.1 fps past the specimens. The results of the tests indicated that the corrosion rate of Ag-In-Cd under the above conditions is a straight line function.

The results of the corrosion test of Ag-In-Cd control rod material plus 2% Sn exposed to the preceding conditions (no boric acid) indicated a very slight exposed weight loss for a 30 day exposure.

A specimen of Ag-In-Cd control rod material electrolytically coated with a defected 2 mil nickel plate indicated no peeling of the plate after 2 weeks of testing in 600°F water containing 1 ppm of Li as $LiOH$.

Calculations were performed to determine the induced radioactivity due to corrosion of silver-indium-cadmium control rod material.

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. All data have been analyzed, and completion of the final report on this work is scheduled for the month of September.

3.5 Decontamination and Waste Disposal

Topical report YAEC-90, "A Study of Decontamination Agents for Use in the Yankee Reactor", was completed in draft form.

A decontamination study was begun making use of contaminated stainless steel pipe from the original SLW steam generator.

An exchange of technical information pertaining to the decontamination project was held with representatives of the Turco Corporation of Los Angeles, California. Significant information relating to the chemical cleaning art was obtained and samples of Turco products were provided for use in decontamination studies at (W) APD.

3.6 Crud Transport and Deposition

Calculations relating to the crud study side stream section added to PAR Loop "A" have been completed. These consider systems of heat transfer where air, water and natural radiation are used.

The corrosion test "harp" for PAR Loop "A" has been received. Required vent and inert gas purge valves have been added. The harp has been hydrostatically tested at 3600 psig. The harp, with sample holders, is expected to be installed during the next normal shutdown period which will be about October 1, 1958.

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

An analysis was completed to determine whether a stretch-formed and disc-compartmented fuel rod containing UO_2 pellets clad with AISI 304 stainless steel tubing having a 21 mil thickness would withstand a cold hydrostatic test of 3750 psi at 110°F. Favorable results were obtained and tests will be performed to confirm the results of the analysis.

A control rod follower consisting of fuel rods was designed for use in a comparison study with a solid Zircaloy follower. The fuel-containing follower should produce a lower flux peak.

An investigation was begun of the thermal bowing properties of a compromise fuel assembly design. Preliminary analyses indicate that the compromise design will have less bowing than the present definitive design.

4.2 Control Rod Drive Mechanisms

Design of the test stand for the control rod drive mechanism prototype was begun.

The cooling requirements for the prototype drive mechanism were calculated.

The specifications for the prototype drive mechanism power supply were written and were released for bids.

4.3 Design of Core Support Structure and Fuel Handling Tools

The analysis of the experimental data of the stress and deflection tests on the quarter size model of the baffle was completed. Design changes were made to reinforce the critical deflection areas, and incorporation of these changes to the model is in progress.

The basic design of the hydraulically actuated universal handling tool, including the hydraulic system, was completed.

4.4 Design for Critical Experiment and Irradiation Tests

The fabrication of eight sets of modified cruciform fuel rod supports and lucite insert plates for the CRX 4:1 W/U ratio core was completed.

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 internal pressure of both a non-compartmented and a disc-compartmented fuel rod in the hot channel was calculated.

Preliminary calculations were begun of loss-of-flow accidents in the MTR In-Pile Test Loop.

5.2 Hydraulic Design

The drafts of topical report YAEC-75, "Model Study of the Pressure Drop Relationships in a Typical Fuel Rod Assembly" and YAEC-76, "The Effect of Local Boiling on Pressure Drop and Flow Distribution on the Yankee Reactor Core", were revised.

Hi-speed motion pictures were taken of the action of the control rod hydraulic dashpot prototype. Evaluation of the pictures is now in progress.

6.0 CONTROL ROD DEVELOPMENT

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

6.0 Control Rod Development

Tensile and creep specimens of Ag-In-Cd were removed after 30 days exposure to simulated Yankee operating conditions in a dynamic loop. Analysis of the test is now being made.

A test program has been started to evaluate nickel plating and copper plus nickel plating on Ag-In-Cd. The suggested purpose of the copper transitional layer is to provide a metallurgical bond through diffusion between the nickel plate and base metal. In this experiment, it is planned to electroplate varying amounts of copper (from zero to 1/2 mil) on Ag-In-Cd. The copper will be diffused into the base metal and the nickel plate by heating the specimens to 550 and 600°C for varying durations of time. A metallographic investigation will be conducted on these samples.

A reverse bend test is planned to help evaluate the adhesion of the plating. The third phase of the test is to determine the corrosion resistance of the diffusion zone. To do this, it is planned to expose a cross-section of each specimen to simulated Yankee operating conditions in a dynamic loop.

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 operational check of the PRESTO computer code for a solution of the steam pressurizer transient response problem has been partially completed.

8.0 PLANT SYTEMS 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 August, 1958.

8.11 Reactor Handling Tools

Redesign of a head gasket and seal ring handling fixture was continued. The redesigned fixture will provide simplified latching and unlatching for the gasket and seal ring mechanisms.

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 August, 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

Control rod worths in the CRX 2.23:1 W/U ratio core were studied with both clean and borated moderators.

The void coefficient of reactivity for the borated moderator was studied. The data are being analyzed.

The temperature coefficient of reactivity in the borated moderator required a variation of technique due to water evaporation and the resulting changes in boron concentration. A thin film of paraffin oil, dispersed over the surface of the moderator water, effectively reduced evaporation losses.

Migration area for the clean core was studied by the partial water height technique. Analysis of this work has not been completed.

Several axial flux profiles were taken using a 1/4 inch fission chamber. Preliminary analysis indicates close agreement with the value for reflector savings given by standard techniques.

Several flux profiles were made using gold foil and fuel rod scanning techniques. The data are being processed through the DARED Code Program on the IBM 704 computer.

A silver-indium-cadmium experimental control rod was inserted in the CRX 2.23:1 W/U ratio core to compare its worth with the worth of standard silver-cadmium control rods. The worth of a second silver-indium-cadmium control rod, of smaller dimension, was compared with the worth of a hafnium control rod of the same size in both a clean and a borated moderator. Detailed analysis of the data is in progress.

The second draft of a YAEF topical report covering measurements on the CRX 3:1 W/U ratio core is nearing completion.

On August 22, 1958, the USAEC issued Amendment No. 2 to Westinghouse Electric Corporation Operating License No. CX-6. This amendment permits the final series of Yankee Critical Reactor Experiments with a 4:1 W/U ratio core to be performed.

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 assembly of the test loop was completed with the exception of the installation of two water sampling points in the purification system and provisions for loop drains which will connect with the MTR drainage system.

Preparations for the functional test of the WCAP-4 test loop were made at the Lummus Company in Newark, New Jersey. Chemical equipment was supplied and a Lummus technician was trained to perform the chemical tests required to assure compliance with the test conditions specified.

The loop instrumentation was delayed approximately two weeks due to late deliveries.

Flow data were obtained which substantiates the decision to operate with two pumps with a third pump as a stand-by. Two pumps will produce the required water velocity.

The loop was hydrostatically tested, and the necessary corrections and adjustments were made.

11.2 Performance of Radiation Damage Experiments

MTR Flux Data

Capsule WCAP-2-4 has been received by KAPL for post-irradiation examination. WCAP-2-4 was discharged from position A-27-NW at the end of cycle 107, August 4, 1958, with a total of 8500 MWD/T and an accumulated, unperturbed thermal nvt of 7.7×10^{20} .

Following is a record of the irradiation cycles and corresponding positions of this capsule:

<u>Cycle</u>	<u>Position</u>
94-101	A-14-NE
102-107	A-27-NW

The accumulated, unperturbed thermal nvt for WCAP-2-2, -2-5, and -2-7 as of August 4, 1958 (the latest available information)

is as follows:

<u>Capsule</u>	<u>Accumulated Unperturbed nvt</u>
2-2	7.3×10^{20}
2-5	7.5×10^{20}
2-7	7.1×10^{20}

Post-Irradiation Examination of WCAP-1 and -2 Experiments

KAPL is continuing their analysis of the first and second groups of samples (WCAP-1-1, 1-3, 1-5, and 2-1) and (WCAP-1-2, 1-4, 2-3, and 2-6) respectively, that have been removed from the MTR.