MASTER
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YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

MONTHLY PROGRESS REPORT

OCTOBER, 1958

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

NOVEMBER 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-101

MONTHLY PROGRESS REPORT

for the period

October 1 to 31, 1958

by

R. W. Garbe
H. E. Walchli

Large Plant Engineering

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

November 20, 1958

APPROVED:

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A. E. Voysey
Project Manager

Westinghouse
ELECTRIC CORPORATION
ATOMIC POWER DEPARTMENT
P.O. BOX 355
PITTSBURGH 30, PA.

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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 October, 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 October, 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 describe the work accomplished from the beginning of the program, June 6, 1956 to September 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 following investigations of fuel fabrication procedures on a production scale at the Westinghouse Materials Manufacturing Department were continued:

1. Reductions in the cost of pellet density determinations
2. Elimination of pellet hour-glassing
3. Relationship between poor granulation and the occurrence of laminae
4. The effect long sintering times and high temperatures have on the manifestation of laminae in pellets which had low green densities.

Attempts were made to ascertain the source of water vapor found in some of the irradiated process water fuel capsules by air drying pellets after centerless grinding, weighing, drying for 72 hours at 80°C and re-weighing.

Pelletization behavior trials were conducted on two lots of UO_2 powder using 10 material preparation methods and various pressing pressures. Centerless grinding of the resultant pellets exposed subsurface laminae of varying degrees of severity. The lack of a draft on the die for the semi-automatic pellet press was considered to be the cause of the laminae.

1.3.1 End Closure of Fuel Rods

Preparation was begun of the draft of a topical report describing the development of methods and use of materials employed in the end-closure of stainless steel fuel rods.

1.3.2 Joining Fuel Bundles into Assemblies

A 36-tube, disc-compartmented experimental fuel subassembly 60 inches long containing Al_2O_3 pellets and having 21 mil thick, AISI 304 stainless steel, Weldrawn cladding was brazed at 1875°F for three hours. The tube spacer ferrules were plated with Ni-P brazing alloy. After collapsing studies are concluded the subassembly will be sectioned, checked for tube-to-tube spacing and tested for joint strength.

Attempts were made to eliminate the brittle Ni_3P phase from brazed joints by extended heat treatment. Examination of the microstructures of joints held for 24 hours at 1900°F showed the presence of the brittle phase, indicating that removal of the brittle phase of heat treatment was unsuccessful.

Free-gap length and percentage elongation data were obtained from the stretch-forming of a full-length Yankee type fuel rod loaded with depleted UO_2 pellets. The fuel column length consisted of 150 pellets divided at 15 pellet intervals into 10 compartments.

1.4 Fabrication of Fuel Elements for Critical Assembly and Irradiation Tests

Fabrication of the remaining process water and in-pile test loop samples was completed. The six process water samples were shipped to the MTR. The in-pile test loop samples are being held for shipment to the Lummus Company until they are required for insertion in the out-of-pile section of the test loop. The nickel-plated silver-indium-cadmium control rod alloy samples are available for shipment to the Lummus Company for insertion in the in- and out-of-pile sections of the test loop.

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 October, 1958.

2.2 Core Steady State Analysis

A study is in progress to compare calculated and experimental neutron flux distributions in water slots in the Yankee CRX cores. Good agreement was obtained between experiments and calculations involving two group diffusion theory, hand calculated fast constants and SOFOCATE calculated thermal constants. Comparisons are in progress utilizing transport theory and fast group constants calculated by means of the MUFT IV code for the IBM-704 computer.

Work was begun on the application of nuclear design procedures, fitted to the Yankee critical experiments, to Yankee core designs.

A study using the CANDLE burnout code has shown that the rod follower power peak decreases with burnout of the surrounding core.

2.3 Core Kinetic Analysis

An analog computer study of transient conditions during startup of the Yankee reactor was completed. The data obtained are being analyzed and a report is being prepared.

A procedure was established for the study of loss of flow transients by iterative application of the flow redistribution program for the IBM-704 digital computer and the analog computer simulation procedure previously developed for the study of this problem.

2.4 Control Rod and Chemical Poison Analysis

The PDQ computer program was applied to determine the detailed power distribution in a cross section of the axial region of the Yankee core containing the point of maximum power density. The results of this study permit a more accurate evaluation of the ratio of peak power to average power in the reactor. The ratio of peak power to average power in a typical two region core was also determined, and the local flux peaking due to control rod followers was investigated.

A study was made of control rod programming in a two region power flattened Yankee core.

2.5 Critical Experiment - Planning and Analysis

Criticality values obtained by use of assumed values of resonance escape probabilities and flux disadvantage factors with the CANDLE code were found to be in good agreement with experimental criticality values for the three Critical Reactor Experiment cores.

Experiments designed to provide total worths of banked control rods have provided high values (28% in the 3:1 w/u ratio core, 25% in the 4:1 w/u ratio core) confirming the importance of fast neutron absorptions previously predicted. Studies indicate that the much lower values arising from measurements of individual rod worths can be attributed to large positive interactions between control rods.

2.6 Irradiation Experiment - Design and Analysis

Nuclear physics calculations for the MTR in-pile test loop experiment were prepared.

2.7 Shielding Analysis

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

2.8 Startup Experiment Assistance

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

Resin capacity tests were performed in the dynamic autoclave purification system using Rohm and Haas XE-150 (H-OH) resin without filters, as part of the purification system resin evaluation program. The amounts of and reasons for the concentrations of iron, nickel, chromium, molybdenum and copper on the resin were determined.

3.3 Corrosion of Materials of Construction

Silver-Indium-Cadmium Corrosion Test Program

A program was established to investigate the corrosion resistance of nickel and nickel-copper plated silver-indium-cadmium control rod alloy under simulated Yankee steady state conditions of high temperature and pressure, at both high and low velocities.

✓ Corrosion testing of plated silver-indium-cadmium control rod alloy samples in (W) APD Loop "A" was completed. The nickel plating protected the samples and no attack on the nickel plate was evident after one month's exposure to water having a pH of 10 by the addition of LiOH, at 608°F, 1800 psi with H₂ overpressure, and a flow rate of 38 fps. Two samples having intentionally defective nickel plating showed that the exposed silver-indium-cadmium was not protected electrochemically by the presence of the nickel plate. The exposed metal in the defected area corroded at the rate of the unplated blank.

✓ The control rod alloy samples in the dynamic autoclave test have been exposed for 14 days to water having a pH of 10 by the addition of LiOH at 600°F, 1525 psi, with H₂ overpressure and a flow rate of 0.1 fps. The unplated samples in the test have shown expected initial corrosion rates of about 60 mg/dm²/month.

Simulated Reactor Startup and Shutdown Tests

A dynamic autoclave corrosion test to determine the effect of concentration cycling of boric acid on corrosion of Yankee primary system materials is continuing after 14 days exposure to water with a pH of 5.8 to 10 controlled with 1 ppm Li as LiOH. Another test in water with a pH of 5.8 to 7 (no Li added) was terminated after 28 days due to a large deposit of silver-cadmium on all specimens except Inconel and Microbraz-50.

No cause for this phenomenon was apparent, however, the Bettis Plant has reported weight losses for 80Ag-15In-5Cd alloy that approximated the amount observed in this test for a solution containing a high oxygen level. An attempt to duplicate the weight loss by exposing similarly preconditioned Ag-In-Cd specimens to a high (6 ppm) oxygen environment was unsuccessful.

Revision of Data for Topical Report YAEC-68

A revision of the data for the final draft of topical report YAEC-68 "Corrosion of AISI 304 Stainless Steel in High Temperature Borated Water" and the subsequent revaluation of the results were completed. The chemical descaling solutions and techniques used to obtain the new descaled weight values satisfactorily removed the corrosion scale from all of the test coupons.

Reaction of Soluble Oxygen with Hydrazine in Borated Water

Completion of the test involving the reaction of soluble oxygen with hydrazine in borated water has been delayed pending the resolution of difficulties encountered in analytically determining the oxygen content of the water.

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

Preparation of the final draft of topical report YAEC-93 "Corrosion Product Transport and Deposition Under Ionizing Radiation" is continuing.

3.5 Decontamination and Waste Disposal

Preparation was begun of a topical report entitled "Development of Agents and Procedures for Decontamination of the Yankee Reactor Primary Coolant System."

The series of dynamic decontamination experiments using contaminated tubing sections from the original SLW steam generator was completed.

3.6 Crud Transport and Deposition

The study to determine crud deposition as a function of heat flux was continued. Sections of the heat exchanger from the Van de Graaff crud deposition experiments are being used in the test. Calculation of heat flux as a function of heat exchanger length have been prepared and will be compared with the amount of deposited crud found on incremental segments of the heat exchanger piping.

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

Hydrostatic pressure testing of various types of stainless steel tubing is continuing. Preliminary results indicate that AISI 304-L, 304-L stretch-formed, and 304 stretch-formed and annealed 1875°F for 4 hours, collapse at a pressure substantially lower than 3750 psig at 100°F. AISI 304 and 316 tubings annealed and not stretch-formed showed that they can withstand a 3750 psig pressure at a temperature up to 125°F.

4.2 Control Rod Drive Mechanisms

Fabrication of the test stand for the positive engagement type control rod drive mechanism is near completion. Work was begun on the fabrication of the control console and drive assembly test fixture. The control console contains the position indicators and automatic programming equipment. The test fixture will be used for checking alignment and operation of the drive assembly.

4.3 Design of Core Support Structure and Fuel Handling Tools

The first draft of a topical report entitled "Stress and Deflection Analysis of the Yankee Core Baffle Structure" was completed.

Fabrication of the Yankee Universal Handling Tool in the Westinghouse APD Model Shop is continuing. It was originally proposed that actuation of the tool would be accomplished by use of a hydraulic power pack mounted on top of the manipulator boom, however, the design was changed, and the tool is now being fabricated for operation by compressed air.

4.4 Design for Critical Experiment and Irradiation Tests

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

Process Water Specimens

The maximum heat flux and corresponding temperatures and temperature drops were calculated for the WCAP-2-4, -5, and -7 process water specimens; the calculations were based on the following assumptions:

1. There is a gap between pellet and cladding which is filled by a gaseous mixture.
2. The composition of the gas is 85% Xe and 15% Kr.
3. The gas pressure is either 5830 or 3180 psi.

The temperatures calculated for the center of the fuel vary with the gas composition and pressure. The range of the center of fuel temperature was calculated to be between 4800°F and 5900°F.

In-Pile Test Loop

The hazards analysis on the loss of flow accident for the in-pile test loop was written and included in the WCAP-4 In-Pile Test Hazards Report.

Internal Pressure of Fuel Rods

The study of gas pressure in the fuel rod was extended to show the effect of the cold axial gap upon the final internal pressure.

Fission Gas Release

Fission gas release from the pellets was calculated as a function of the nuclear hot channel factor to determine the effect of hot channel factors on the internal gas pressure.

Calculations performed by the Westinghouse APD Nuclear Design Section for the fission gases released from pellets were checked. A literature investigation was begun to obtain more information on the process of fission gas release from UO₂ pellets.

5.2 Hydraulic Design

Ferrule Design Fuel Assembly Hydraulic Tests

Work was initiated on the testing of a ferrule design fuel assembly in a hydraulic loop. Components were fabricated, and a jig was built for assembling the desired ferrule arrangement.

The pressure drops across the definitive, compromise, and individually supported fuel rod fuel assembly designs were calculated and compared.

Reactor Vessel Model

The reactor vessel plastic model was modified for additional hydraulic testing. A vessel head model was fabricated to simulate actual reactor head conditions. A special manometer assembly was built to aid in velocity gradient studies.

Control Rod Dashpot Tests

Dashpot hydraulic tests were given additional impetus with the completion of an accelerometer and recorder setup. Use of the accelerometer assembly will make it possible to obtain direct acceleration versus time data. As a consequence of the latest available test data, holes were drilled in the pistons to remove all air. New pistons designed to complete the experimental data are now being fabricated. Completion of the tests on the hydraulic dashpots are scheduled for the middle of November.

Preparation was begun of the draft of a topical report entitled "Hydraulic Shock Absorbers for the Yankee Reactor Control Rods."

6.0 CONTROL ROD DEVELOPMENT

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

6.0 Control Rod Development

Preparation of ordering information has been initiated for the procurement of silver-indium-cadmium alloy from KAPL which will be used to fabricate a prototype control rod.

The corrosion testing of plated and unplated silver-indium-cadmium control rod alloy samples is reported in subproject 3.3 under "Silver-Indium-Cadmium Corrosion Test Program."

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 preliminary draft of the analog computer study for the Yankee primary plant control system was completed. However, due to system parameter changes and incomplete data, a second analog computer study is presently being run, the results of which will be combined with the original study. The purpose of these computer studies is to develop a method of controlling the reactor to limit the transients of the system.

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 October, 1958.

8.11 Reactor Handling Tools

A study was made of the problems that would be encountered in the storage and handling of large reactor internals (guide tube support and hold-down plates and upper core support barrel) in the shield tank cavity.

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 October, 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

4:1 Water/Uranium Metal Core

Measurements of the radial and axial buckling and reflector savings have been completed, but the data on the axial measurements have not been analyzed.

Peripheral fuel rod worth measurements were performed in a clean core of 25.3 cm radius, 1330 fuel rods, and five control rod followers and in a borated core containing 780 ppm boron, having a radius of 46.6 cm, 4500 fuel rods, and with nine control rod followers.

Measurements were made on the resonance escape probability in a clean and in a borated core.

Similar sets of data were obtained leading to values for the fast fission factor. Values of the ratio of activity between enriched and depleted foils were determined when measured approximately 10,000 seconds after irradiation. Malfunctioning of a scaler led to large probable errors in the measurements.

A value was found for the disadvantage factor in the 4:1 water/uranium ratio core.

Measurements of partial water height leading to values of the migration area have been made. Results obtained have not been analyzed.

The above experiments successfully completed the Yankee Critical Reactor Experiments. The experiments were conducted without incident, and a substantial amount of very valuable information was obtained. A topical report covering the experiments with 2.23:1, 3:1 and 4:1 water/uranium ratio cores is being written.

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

Repairs and modifications were made to the in-pile test to correct malfunctioning observed during a preliminary operational test at the fabricator's plant. Many of the mechanical items require corrective action before the loop can be subjected to operating conditions for the final acceptance tests. The items which require modification prior to test operation were analyzed and corrected except for the pressurizer heater replacement. The heater replacement is scheduled for completion by November 11, 1958. It is planned to run a 24 to 36 hour operation test upon completion of the modifications.

Preparation of the Operating Manual was completed.

The Hazards Analysis Manual is being given a final review.

A preliminary draft of the Preliminary Maintenance Manual has been sent to the MTR for their review.

Westinghouse APD is awaiting receipt of the Vendors' Manuals for test loop components and the Erection Procedure Manual from the Lummus Company.

11.2 Performance of Radiation Damage Experiments

Irradiation of Process Water Capsules

The 30-inch long process water sample shipped to the MTR with five other capsules was inserted in the MTR. The remaining five process water samples will be inserted in the Experimental Test Reactor instead of in the Materials Testing Reactor because of their high thermal flux requirements.

Process water capsules WCAP-2-2, 2-5 and 2-7 have received the following accumulated unperturbed nvt at the end of the MTR cycle 111:

<u>Capsule</u>	<u>Accumulated Unperturbed nvt</u>
WCAP-2-2	9.8×10^{20}
WCAP-2-5	10.0×10^{20}
WCAP-2-7	9.4×10^{20}

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

Process water capsule WCAP-2-4 has not been prepared for examination by KAPL due to renovations being made in their hot cell facilities. Examination of the capsule is expected to be begun about the middle of November.