WCAP-3269-6
UC-80 Reactor Technology
Special Distribution

MASTER

LARGE CLOSED-CYCLE WATER REACTOR RESEARCH AND DEVELOPMENT PROGRAM PROGRESS REPORT

FOR THE PERIOD

OCTOBER 1 TO DECEMBER 31, 1964

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Project Manager

Prepared for the U. S. Atomic Energy Commission
Under New York Operations Office Contract AT(30-1)-3269

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1.00 PROJECT ADMINISTRATION
C. Roderick, Project Manager

Overall Coordination
C. Roderick, R. W. Garbe

AEC approval was received for the revised Work Program for the July 1 to December 31, 1964 Period incorporating changes to existing sub-task work programs and additional sub-task work programs describing the development of the advanced grid, canless rod cluster control fuel assembly concept.

The Work Program for the January 1 to June 30, 1965 Period was prepared and submitted to the AEC for approval.

LRD Program Reports

Direction and coordination of the preparation of quarterly progress reports and topical reports for the Research and Development Program was continued. The quarterly progress reports without an asterisk listed in Table 1.1 have been issued to the AEC Division shown on the External Distribution List. The quarterly progress reports marked with an asterisk are being reviewed by the AEC.

Table 1.1 - LRD PROGRAM REPORTS

<u>Report Number</u>	<u>Quarterly Progress Reports</u>
WCAP-3703	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period December 1, 1960 to May 31, 1961
WCAP-3706	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period June 1 to August 31, 1961
WCAP-3707	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period September 1 to November 30, 1961
WCAP-3708	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period December 1, 1961 to February 28, 1962
WCAP-3710	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period March 1 to June 30, 1962
WCAP-3718*	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period July 1 to September 30, 1962

Table 1.1 (Continued)

<u>Report Number</u>	<u>Quarterly Progress Reports</u>
WCAP-3721*	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period October 1 to December 31, 1962
WCAP-3738*	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period January 1 to March 31, 1963
WCAP-3739*	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period April 1 to June 30, 1963
WCAP-3743*	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period July 1 to September 30, 1963
WCAP-3750*	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period October 1 to December 31, 1963
WCAP-3269-2	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period January 1 to March 31, 1964
WCAP-3269-3*	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period April 1 to June 30, 1964
WCAP-3269-5*	Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period July 1 to September 30, 1964

The following Topical Reports have been approved by the New York Operations Office and issued in accordance with the External Distribution List.

<u>Report Number</u>	<u>Topical Reports</u>
WCAP-1570	Literature Values for Selected Chemical/Physical Properties of Aqueous Boric Acid Solutions
WCAP-1689	The Behavior of Austenitic Stainless Steel Corrosion Products in High Temperature Boric Acid Solutions
WCAP-1700	Ion Exchange with Equilibrium and Decay
WCAP-1831	Primary System Pressure Response Study Due to Power Unbalance
WCAP-1997 (Rev. 2)	New DNB (Burnout) Correlations
WCAP-2048	The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements

Table 1.1 (Continued)

<u>Report Number</u>	<u>Topical Reports</u>
WCAP-3701	Design and Hazards Report, LRD Irradiation Test Number 1
WCAP-3704	"THINC" A Thermal Hydrodynamic Interaction Code for a Semi-Open or Closed Channel Core
WCAP-3709	Compilation of a MJFT Library (Library 567)
WCAP-3711	Heterogeneous Representation of Control Rods in Light Water Lattices - A Comparison of Analysis and Experiment
WCAP-3713	Some Physiochemical Studies of Boric Acid Solutions at High Temperatures
WCAP-3714	Reactor Control and Protection System Study for Chemical Shim Operation of a Large Closed-Cycle Water Reactor
WCAP-3716	Ion Exchange in Boric Acid Solutions with Radioactive Decay
WCAP-3717	RENEWED: A Program for Nuclear Reactor Fuel Cycle Analysis
WCAP-3719	Control Rod Programming in a Large Multiregion Core
WCAP-3720	Reactivity Dependence Upon Moderator Temperature and Boron Concentration in Large PWR Cores Controlled by Chemical Shim
WCAP-3722	A Digital Computer Analysis of Closed-Cycle Water Nuclear Power Plant Startup Using Natural Circulation (THERMO-SYPHON Code) Volumes I and II
WCAP-3723	Fine Structure Power Peaking in a Critical Experiment Mockup of a Chemical Shim Core
WCAP-3724	Load Variation Restrictions in a Chemically Poisoned Large PWR Core
WCAP-3727	Control Rod Requirements in a Chemical Shim Core
WCAP-3729	The In-Pile Test of Chemical Shim
WCAP-3730	Interactions Between Stainless Steel Corrosion Products and Boric Acid Solutions
WCAP-3731	Radiotracer Studies of Hideout at High Temperature and Pressure
WCAP-3733	Optimization Study for Large Pressurized Water Reactor Cores
WCAP-3734	Fabrication and Irradiation of Stainless Steel Clad UO_2 Fuel Rods
WCAP-3735	Coolant Mixing in Open Lattice Reactor Cores
WCAP-3737	A Long Term Corrosion Test of Reactor Structural Materials in a Chemical Shim Solution

Table 1.1 (Continued)

<u>Report Number</u>	<u>Topical Reports</u>
WCAP-3269-8	Hydraulic Tests of the San Onofre Reactor Model
WCAP-3269-25	Calculations of Lattice Parameters and Criticality for Uniform Water-Moderated Lattices
WCAP-3269-27	Saxton Kinetic Experiments

2.00 REACTOR PHYSICS DEVELOPMENT

H. W. Graves, Jr., Manager

Nuclear Engineering Section

R. J. French, Manager

Nuclear Design Section

G. H. Minton, Manager

Reactor Physics & Mathematics Section

201 Light Water Reactor Core Physics Calculation Methods

W. J. Eich

With the completion of the report¹ described under Subtask 7.01, some effort was expended on criticality studies. In particular, values of thermal utilization for room temperature criticals were re-calculated by means of the THERMOS code in order to reflect minor improvement in the microscopic library cross sections. This effort is a necessary preliminary to an over-all study aimed at resolving an eigenvalue discrepancy related to pellet size.

Criticality calculations in the past had been extended so as to include, in addition to room temperature criticals, two borated configurations of the hot Yankee reactor. Subsequent to this extension, some important modifications were made in the methods used in the determination of critical eigenvalue. These changes, based on analysis of room temperature criticals, will be incorporated in the description of two Yankee configurations. This particular effort, however, is much more involved than the analysis of room temperature critical assemblies because of the many different heterogeneities in the Yankee reactor. The initial work on this revision of the representation of Yankee was carried out.

202 Local Power Disturbances

L. E. Strawbridge, R. F. Barry, W. D. Leggett

The preparation of topical report WCAP-3269-48 "Xenon-Induced Core Instabilities" was begun. The report summarizes the results of the xenon oscillation studies performed under the Large Closed-Cycle Water Reactor Research and Development Program.

The results of the overall studies indicate that xenon oscillations are not expected to occur in the San Onofre Nuclear Generating Station since the negative Doppler coefficient provides strong damping. Furthermore, it has been shown that even if oscillations could occur in the uncontrolled reactor, they could be prevented by appropriate use of the in-core instrumentation and the reactor control system.

¹ WCAP-3269-38 "The Analytic Representation of Rod Cluster Control Elements in Nuclear Design Calculations" (to be published).

203 Nuclear Design - Fuel and Control Management
D. L. Miller, W. T. Sha

The preparation of topical report WCAP-3269-47 "Reactivity Worth of the Rod Cluster Control Elements" was begun. A large number of measurements of the worth of rod cluster control absorber elements have been made for different numbers, arrays and sizes of absorber rods. An analytical method of evaluating the worth of RCC elements was presented in WCAP-3739 "Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period April 1 to June 30, 1963". Additional experimental data has been incorporated into the results of the previous studies and summarized in topical report WCAP-3269-33 "Control Rod Worth in a Heterogeneous Mockup of a Large PWR Core" which describes the analytical methods and compares the calculated rod worths to the measured worths.

205 Development of Nuclear Instrumentation System
J. J. Loving

A second long ionization chamber was installed at Carolinas-Virginia Tube Reactor for experimental data-taking, with the first one being transferred over for use in one of the operational power range channels. Hourly data are being logged of the readings of these two chambers, along with measured reactor and steam generator thermal powers, and control rod positions. Analysis of the data will be made during the next quarter to determine the reduction in errors possible by using long chambers instead of normal size ones.

A purchase requisition was placed for another prototype long chamber with two equal internal axial volumes, one uncompensated and the other compensated for gamma response, to be fabricated by Reuter-Stokes Electronic Components, Inc. The use of this type would allow some measure of reactor axial flux tilts by indicating the currents from the two sections separately, and using the total current for reactor protection and alarms. At the same time, the compensated sections would eliminate the need of separate detectors for use in the intermediate range or for physics tests.

3.00 REACTOR CORE - DESIGN AND FABRICATION DEVELOPMENT

R. J. Allio, Manager Materials and Processes Development
A. G. Thorp, Manager Reactor Engineering
P. Cohen, Manager Chemical Development

301 Development of Advanced Fuel Assembly Fabrication

R. Duncan

The preparation was begun of WCAP-3269-37 "A New Zircaloy End Closure Weld Joint Configuration" describing the newly developed method of seal welding end plugs to Zircaloy tubing which reduces the number of welding operations and minimizes weld buildup.

301.1 Mechanical and Physical Properties of Zircaloy

D. B. Scott W. R. Smalley

Tests were continued to determine tangential mechanical properties of Zircaloy tubing. Repeated attempts to determine 0.2% offset yield strength have been invalidated because of strain gage cement failure at stress values below the 0.2% yield strength. To alleviate this problem, a circumferential extensometer of the type developed under LRD 301.7 is being machined.

Results of burst tests with zero axial stress, run at 750°F and 800°F, are given below. Axial ultimate strengths are shown for comparison.

<u>Temperature</u>	<u>0.2% Offset Yield</u>	<u>Tangential UTS, psi</u>	<u>Axial UTS, * psi</u>
680°F*	---	55,600	47,600
750°F	---	52,000	40 to 45,000
750°F	---	49,400	40 to 45,000
800°F	36,400	43,200	34,000

*Reported previously

Both fractures at 750°F can be described as a torsion type of break, i.e., the break was in the shape of a helix and little ductility was indicated. It is postulated that this may be due to residual torsional stress inadvertently applied from fastening the swagelock fittings. The break at 800°F was ductile as was the break at 680°F reported previously. A strain gage remained attached on the sample tested at 800°F so the yield strength was determined.

It is expected that the tests to determine tangential properties of Zircaloy cladding will be completed during the next quarter. In addition, the preparation of WCAP-3269-41 "Physical and Mechanical Properties of Zircaloy-4 Fuel Cladding" will begin.

301.2 Hydriding Studies of Zircaloy Fuel Element Cladding
K. C. Thomas

The purpose of this program is to study and evaluate the effect of heat transfer and isothermal conditions on the corrosion and hydrogen pickup of Zircaloy cladding in a pressurized water environment. The results of this study will be used to determine the limits to which Zircaloy cladding can be utilized without degradation of mechanical properties due to hydrogen pickup.

The first test run was started on December 17, 1964 under the following conditions:

Surface temperature	680°F
Bulk water temperature	600°F
Nucleate boiling in one section:	forced convection in one section
Water chemistry:	1400 ppm boron as boric acid
	4 ppm potassium as KOH
	<1/10 ppm Cl
	<1/10 ppm O
	30 cc H/Kg
Test time:	15, 70, 120, 200 days

The second facility is nearing completion and will be placed under test as soon as some modifications are made to the heater section. After the shakedown run on Facility #1 it became apparent that the excess heat loss from the autoclave walls would not allow tests to be run at 640°F bulk water temperature. This limitation has been overcome by the installation of an auxiliary heater in the system. A similar modification will be made to Facility #2.

301.6 Fabrication of the Fuel Assembly R. Duncan

The components for two prototype canless fuel assemblies were manufactured. The final assembly of the prototype for evaluation in Loop D was completed and released to the Mechanical Design Group. While the prototype for the mechanical evaluation will be completed in early January 1965 many manufacturing techniques had to be developed and refined.

The tooling, fixturing and brazing techniques had to be developed for grids made from Inconel 718. Eighteen grids were manufactured. In developing a brazing procedure many variables were investigated. These included Microbraz 50 and Microbraz 30 as brazing alloys; vacuum, hydrogen and argon as the brazing atmospheres; and a nickel plate, a $\text{HNO}_3 + \text{HF}$ pickle of Turco 4101 as surface preparation of the Inconel 718 grid straps.

The grids that were nickel plated and brazed in Hydrogen using Microbraz 30 as the brazing alloy appeared to give the best braze joint. All the nickel plate on the grids had to be stripped off with a $\text{HNO}_3 + \text{HF}$ pickling solution before being assembled into a can.

One hundred eighty dummy fuel rods were fabricated for use in the prototype tested in Loop D. Four types of fuel rods were represented as shown below.

- 60 Pre-corrosion tested Zirc-4 rods
- 10 As-pickled Zirc-4 rods
- 6 Burnished Zirc-4 rods
- 104 Type 304 stainless steel rods.

Lead pellets were used to simulate the UO_2 fuel pellets in all rods tested. An out-of-box welding technique was developed and used to make the final end closure of the 60 pre-corrosion tested Zircaloy-4 rods. In this technique the welding is performed in a small purge chamber. This purge chamber used an O-ring at the chuck and a boot type at the welding torch and allowed the helium to pass out the 0.500 I.D. 6 inch long tube so that the Zircaloy-4 fuel rod was extending through. All the welding appeared to be of good quality and showed no evidence of discoloration.

Sixteen Type 304 stainless steel sheath tubes were fabricated using standard manufacturing procedures. However, special techniques had to be developed to weld the 0.012 inch thick sheath tubes to the upper end plates and in welding the sheath tube to the grids. The sheath tube was hand TIG welded to an adaptor that acts as an 0.020 inch bushing through which the end of the sheath tube passes. A copper chill was placed inside the sheath tube and the exposed end of the tube and adaptor was welded using a hand TIG technique with a current of 8 to 10 amps and a helium atmosphere.

A fixture was designed and built to hold the grids perpendicular to the sheath tubes during fabrication of the skeleton. The fixture is a 12 foot long angle with brackets boxing in the grids at each grid location. Tabs extended from the corner of the brackets thus preventing any movement.

TIG spot welding was selected to fasten the grids to the skeleton. The welding parameters used are as follows:

- Current - 24 amps
- Arc Gap - .030
- Time - 20 cycles
- Atmosphere - He 10 CHF
- A copper chill is placed under the weld

Biaxial Strain Fatigue of Cladding Materials

R. E. Schreiber

The primary purpose of this study is to determine the biaxial strain fatigue limits for 10-15% cold worked Zircaloy-4. The program will evaluate the effects of strain range, hold time under load, and biaxial stress state on the fatigue life (number of cycles to failure) at 715°F the maximum normal operating temperature of the clad.

Results of Fatigue Tests

Two tests using type 2¹ specimens were completed early in December. This type of specimen gives an almost pure hoop stress condition. The first specimen failed after 28 cycles and the second after 110. In each case, a longitudinal wrinkle developed. The strains measured during the test are being analyzed. The radii of curvature of the wrinkles will be determined from photographs of the mounted specimens. The radii will be used to calculate the strain due to bending at the wrinkle. In each case, failure was by cracking at the ridge of the wrinkle for a distance of about 0.3 inch.

The four, type 3 specimens (equal biaxial stresses) were tested together for 10 cycles before equipment failure forced shutdown. Later, two of these specimens were tested to 500 cycles without failure. These tests have now been interrupted in favor of running type 1 specimens. All of the type 3 specimens will eventually be tested to failure.

The two type 1 specimens (ratio of principal stresses is 2:1) are being mounted in the autoclave heads and will be tested early in January.

Design Change in Specimens

A new drawing of the type 2 specimen was prepared to show the specimen which was actually made. The outer bellows housing was omitted. The actual type 2 specimen is shown in Figure 3.1.

¹See WCAP-3750 "Large Closed Cycle Water Reactor Research and Development Program Progress Report for the Period October 1 to December 31, 1963" for descriptions of specimen types.

3.6

BILL OF MATERIAL						NO. REQ.	
ITEM	DESCRIPTION	QUANTITY	UNIT	MATERIAL SPECIFICATION	REFERENCE	QTY	QTY
1	HOUSING	1	PC	10-24 TAP THEN		1	
2	GUIDE	1	PC	10-24 TAP THEN		1	
3	MALE CONNECTOR	1	PC	10-24 TAP THEN		1	
4	MALE CONNECTOR	1	PC	10-24 TAP THEN		1	
5	TUBE	1	PC	10-24 TAP THEN		1	
6	ADAPTOR	1	PC	10-24 TAP THEN		1	
7	FLANGE	2	PC	10-24 TAP THEN		2	
8	SHAFT	1	PC	10-24 TAP THEN		1	
9	BELLOWS	1	PC	10-24 TAP THEN		1	
10	MALE CONNECTOR NO. 12-8 FRTX-96	1	PC	10-24 TAP THEN		1	
11	TUBE	1	PC	10-24 TAP THEN		1	
12	100-32 R. 188LG PLAT POINT SEC SETS	2	PC	COMM 18-B		2	
13	RETAINER	1	PC	10-24 TAP THEN		1	
14	10-24 TAP THEN SEC. HS. CAP SEC.	2	PC	COMM 18-B		2	

A- B.O. FROM PARKER-HANNIFIN CORP CLEVELAND, OHIO.

NOTES:

1. SLOT IN IT. 7 IS SHOWN POSITIONED FOR INITIAL INSTALLATION OF ASSEMBLY. ITEM 7 MUST BE POSITIONED SO AS TO LOCK ASSEMBLY IN PLACE AFTER FINAL ADJUSTMENT.

2. WELD PER W P S 292613-1
3. BRAZE PER W P S 252713-1

Figure 3.1

High Temperature Specimen Fatigue No. 2 Central Section

301.8 Stress Corrosion of Fuel Element Cladding
K. C. Thomas

The study of the failure of stressed Zircaloy-4 in aqueous ferric chloride solutions has now been completed. It was noted that the 0.035 in. diameter wire specimens of Zircaloy-4 and crystal bar zirconium (both in the α -treated condition) were susceptible to failure in solutions of from 5 weight % to 20 weight % ferric chloride at 68°F and a pH of 0.6 when stressed at 30-50,000 psi. The time to failure of Zircaloy-4 containing 14 ppm hydrogen as a function of chloride concentration and stress is shown in Figure 3.2. The time to failure decreases with increase in hydrogen content as shown in Figure 3.3. In all cases the cracks were found to be intergranular.

Based on the results of this study a model for the failure in aqueous chloride solutions of zirconium alloys containing hydrogen has been proposed. Cracks are initiated in the hydride phase of piled up groups of dislocations. Because of the high matrix ductility, these cracks do not propagate into the metal in the absence of a corrosive medium; consequently time dependent failure is not observed. In the presence of a corrosive environment, the propagation through the metal of cracks initiated in the hydride phase at the surface is facilitated by an electrochemical effect, resulting in eventual catastrophic failure. This electrochemical effect is produced by a difference in energy between the matrix and the slip plane over which the piled up dislocations have transversed. This model predicts a failure which is dependent on the amount of hydride present, the orientation of the hydrides and time.

The stress corrosion cracking study of stainless steel in pressurized water doped with chlorides has been modified since thermal expansion in the original straining jigs resulted in a very much lower stress than that originally expected. This has been corrected by changing the jig material from Inconel to stainless steel.

Type 304 stainless steel cold worked-15% and annealed bar samples (7 in. x 3 in.), with a thickness of 0.040-0.050 in. and one zirconium crystal bar sample have been tested under stresses ranging from approximately 10,000-40,000 psi at the chloride concentrations of $\leq 1/10$ ppm, 25 ppm and 50 ppm and oxygen concentration of 50 ppm. Samples were removed after 100 hours at 650F but have not yet been analyzed.

Since some doubt had been expressed about maintaining the oxygen content in the water at the desired value the top can in the autoclave was fitted with means for monitoring the oxygen content. The results indicate that the oxygen content decreases with time and that the test cans will have to be periodically replenished with oxygen. The next series of tests will have facilities for monitoring and replenishing the oxygen.

TIME TO FAILURE OF ZIRCALOY-4 (14 ppm H) IN AQUEOUS FERRIC CHLORIDE AS A FUNCTION OF CHLORIDE CONCENTRATION AND STRESS

Figure 3.2

E. D. SK. 322500-B

TIME TO FAILURE AS A FUNCTION OF HYDROGEN CONTENT OF
ZIRCALLOY-4 AND CRYSTAL BAR ZIRCONIUM AT 50,000 PSI
IN FeCl_3 SOLUTION

E. D. SK. 322497-B

Figure 3.3

Saxton In-Pile Defect Test 3x3 Assembly

A visual examination of the in-pile defect test 3x3 assembly (No. 503-4-24) after irradiation was performed in the Saxton canal. The experiment was irradiated for approximately 1300 effective full power hours in the Saxton chemical shim environment. No increase in primary coolant activity was observed during the test.

The assembly consisted of nine fuel rods, two of which contained a 15-mil-diameter hole machined through the Zircaloy cladding at the position of anticipated maximum heat flux. One of the defected rods had been previously irradiated to a peak burnup of about 9500 MWD/MTU. The second defected fuel rod had not been previously irradiated. Other rods included a previously irradiated Zircaloy clad rod which was not defected, a stainless steel clad rod which was heat treated to produce a controlled degree of sensitization, and five vibratory compacted UO₂ fuel rods clad with both Zircaloy-4 and free standing Type 304 stainless steel.

The examination yielded the following observations and information:

- (1) The general appearance of the defect test 3x3 assembly was satisfactory. All spring clip fingers appeared to be in intimate contact with the fuel rod cladding and no fretting was observed.
- (2) The crud deposition was, in general, very light. As in the case of the RCC 3x3 assembly, (No. 503-4-23) most of the visible crud was observed to be in the high heat flux region between the second and third grid location.
- (3) No indication of attack in the vicinity of the defects was observed.

Transfer of the assembly to the Westinghouse Post-Irradiation Facility hot cells for destructive examination is scheduled for January 1965.

Saxton Special 9x9 and 2x2 Assemblies

The special 9x9 (No. 503-10-1) and 2x2 (No. 503-9-1) assemblies continued to operate satisfactorily as part of the Saxton core. The 9x9 assembly is scheduled to operate as part of the core until mid 1965. An estimated burnup of approximately 15,000 MWD/MTU and a corresponding fast flux exposure (> 1 Mev) of 1.5×10^{21} nvt will be experienced. The 2x2 assembly is scheduled to be moved to the center of the core during the January 1965 shutdown. Calculations, based on the power output of the assembly while in the peripheral location, indicate a peak power output of 25 Kw/ft will be achieved in the center core location. Center melting of the UO_2 fuel is anticipated.

Post-Irradiation Examination of Zircaloy Clad Rods from Saxton RCC 3x3 Assembly

Post-irradiation examination of the Zircaloy clad fuel rods from the RCC 3x3 assembly (No. 503-4-23) was completed. Representative metallographic specimens were prepared of both the autoclave tested and as-pickled Zircaloy clad rods. Hydrogen pickup of the cladding during exposure was also determined. All pertinent data are being collected and the evaluation of performance is in progress.

Performance of Thin-Walled Stainless Steel Cladding in Saxton

A topical report describing the design, fabrication, and post-irradiation examination of the thin-walled stainless steel clad fuel rods irradiated as part of the Saxton core was prepared. The experiment involved the irradiation of 10 fuel rods as part of special 3x3 assemblies No. 503-4-21 and 503-4-22. The rods experienced a maximum peak burnup of 8900 MWD/MTU at a maximum heat flux of 410,000 BTU/hr ft². Careful visual examination of each rod and destructive metallographic examination of the rods representing maximum exposure revealed no evidence of failure or crack formation although the thin-walled clad was subjected to severe plastic deformation.

Fuel Swelling Studies

J. A. Christensen

Dimensional stability of UO_2 under in-pile conditions simulating anticipated large plant environments is under investigation. Parameters which contribute to swelling (burnup, heat rating, fuel density and grain size, and clad restraint) are being systematically studied by a series of capsule irradiations in ETR.

Experimental Design

Design and hazards calculations were completed and incorporated into the final draft of the Experiment Design and Hazards Manual. This was submitted to the ETR Safeguards Committee and approved without dissent or modification.

Basket design was altered to achieve greater mechanical stability. Irradiation space economy and ease of in-cell disassembly are maintained in the new design.

Component Procurement

All non-fissile components (end plugs, sheaths, basket parts, fuel pin hangers) were completed and passed inspection.

Sintered UO_2 fuel fabrication was completed at Cheswick. All 15 lots of pellets (five different enrichments at three different densities) are in storage and are available for capsule assembly as required. Desired maximum densities of 98% T.D. were not attainable either by conventional pressing and sintering or by 2000°C inert gas sintering. The conventional techniques yielded 96.5% T.D. which is acceptable.

Containers and associated hardware needed for fabricating high energy compacted UO_2 were received from Hanford Laboratories. These are being loaded with UO_2 powder and will subsequently be returned to Hanford for compaction. Pellets machined from this material will be irradiated in some of the fuel rods to provide variable grain sizes in high density fuels.

Assembly Fabrication and Delivery

Six capsule-basket assemblies were completed and shipped to ETR to be inserted prior to cycle 69 which begins in mid-January. Included in this group of assemblies are three capsules which will operate at 12 Kw/ft (WAPD-49-A, B and C) and three control capsules (WAPD-49-M, N, and O) which will be irradiated for one cycle only at 12, 18, and 24 Kw/ft respectively. The remaining nine capsules which will operate for extended periods at heat ratings of 18 to 24 Kw/ft, will be completed and shipped prior to 2/1/65. These will be irradiated beginning cycle 70 (about March 1). Table 3.1 lists the irradiation schedule and parameters for the 15 capsules.

Table 3.1 Fuel Swelling Studies - Capsule Irradiation Schedule & Parameters

(Irradiation Parameters Represented)

			Fuel Density Per Cent Theo- retical			Fuel Sheath				Goal Burnup fissions/cm ³ x 10 ⁻²⁰			
Capsule No.	Charge Date	Heat Rating Kw/Ft	90	94	97	Stainless Steel, 304 0.005 in. wall	Stainless Steel, 304 0.015 in. wall	Zircaloy-4 0.024 in. wall	Zircaloy-4 0.065 in. wall	1	4	8	12
WAPD-49-													
A	1/22/65	12		X	X	X	X	X			X	X	X
B	1/22/65	12		X	X	X	X	X			X	X	X
C	1/22/65	12	X	X	X			X	X		X	X	X
D	3/1/65	18	X		X	X		X			X	X	X
E	3/1/65	18		X	X	X			X		X	X	X
F	3/1/65	18	X	X	X		X	X	X		X	X	X
G	3/1/65	18	X		X	X		X	X		X	X	X
H	3/1/65	24		X	X	X	X	X			X	X	X
I	3/1/65	24		X	X			X	X		X	X	X
J	3/1/65	24		X	X	X			X		X	X	X
K	3/1/65	24	X	X		X	X	X	X		X	X	X
L	3/1/65	24		X	X			X			X	X	
M	1/22/65	12	X	X	X	X	X	X	X	X			
N	1/22/65	18	X	X	X	X	X	X	X	X			
O	1/22/65	24	X	X	X	X	X	X	X	X			

302.6 Irradiation Testing of Advanced Fuel Assembly with Improved Grids
D. R. McClintock

Design, fabrication, and installation of the modified 3x3 assembly (No. 503-4-25) with improved RCC grids was accomplished during this report period. The assembly was inserted into a central location of the Saxton Reactor where it is presently operating satisfactorily as part of the core. Fabrication and assembly was performed in accord with Westinghouse assembly drawings #647J111 (Group 6) and #540F650, Figures 3.4 and 3.5 respectively.

The overall configuration of the modified 3x3 assembly with improved grids is the same as previous Saxton special 3x3 assemblies. The design permits access to individual fuel rods for examination or replacement without requiring head removals or destruction of the containing assembly. Changes from previous assemblies were made to the grid material and the design of the spring clip. The grids were fabricated from Inconel 718 to permit precipitation hardening of the fabricated clip fingers to obtain the desired spring properties. The assembly was fabricated with a new braze material, Microbraz 30, which is more ductile than the nickel-phosphorus braze used in the Yankee assemblies.

The fuel-bearing portion of the assembly, illustrated in Figure 3.6, contains nine standard free-standing stainless steel clad fuel rods which were fabricated in accord with Westinghouse drawing #882D141, Figure 3.7. The location and identification of each rod are illustrated in Figure 3.8. As detailed in Table 3.2, the rods are loaded with pelletized UO_2 fuel of a nominal 5.7 w/o enriched U-235. Two of the rods were fabricated in argon gas with the moisture content controlled to 13 and 50 ppm respectively. The cladding of one rod was heat treated to produce a controlled degree of sensitization.

Since the insertion of the improved grid 3x3 assembly coincided with the initiation of the synthetic crud tests in Saxton, the experiment was removed early in December to permit visual examination and sampling of the corrosion product deposited on the removable rods. The results of this effort are reported under subtask LRD 406. Following the sampling operations the assembly was re-installed for continued steady state exposure.

MODIFIED SAXTON 3x3 ASSEMBLY WITH IMPROVED GRIDS

Figure 3.6

BILL OF MATERIAL				AS BUILT									
ITEM	QTY	DESCRIPTION	REMARKS	1	2	3	4	5	6	7	8	9	10
Y 1 PELLET STACK	WES 76 64 1												
2 END PLUG	WES 76 11 1												
3 END PLUG	WES 76 11 1												
4 END PLUG	WES 76 11 1												
5 FILLER	WES 76 11 1												
6 TUBE	WES 76 11 1												
7 TUBE	WES 76 11 1												

X - MOISTURE CONTENT OF FUEL IN GROUP 3 TO BE ADJUSTED ON:
 WTS. BASIS TO FOLLOWING: GROUP 3 - 50 PPM.
 Y - FUEL RODS IN GROUPS 3 & 4 TO BE LOADED AND WELDED IN AN ARGON
 ATMOSPHERE WITH MOISTURE CONTENT AS FOLLOWS:
 GROUP 3 - 50 PPM; GROUP 4 - 15 PPM.

NOTES:

1. WELD PER WPS 292712.
2. INSCRIBE WITH VIBRATING PENCIL OR ELECTRO-ETCH IDENTIFICATION NUMBER ON TOP END PLUG WITH CONSECUTIVE NUMBERS. RECORD IDENTIFICATION NUMBER, FUEL WEIGHT, & METAL WEIGHT.
3. MOISTURE CONTENTS OF PELLETS PRIOR TO WELDING MUST NOT EXCEED 30 PPM ON WEIGHT BASIS EXCEPT AS SPECIFIED IN NOTE 2 ABOVE. TUBE MUST BE DRY IN ALL CASES.
4. PRIOR TO ASSEMBLY, Wipe inside & outside surfaces of tube, spacer, & end plugs (USE ALCOHOL ONLY AS A SOLVENT) WITH SWABS TO REMOVE ALL FOREIGN MATTER. WIPE WITH DRY SWABS AFTER CLEANING.
5. ASSEMBLY MUST BE STRAIGHT WITHIN .000/FT. BETWEEN END PLUG WELDS WHEN LYING ON A SURFACE PLATE. END PLUGS TO BE STRAIGHT WITHIN .010.
6. ADD FILLERS AS NECESSARY TO OBTAIN SPECIFIED DIMENSION (USE 1 FILLER AT BOTTOM OF TUBE ON GROUPS 2, 3, & 4). MEASURE & RECORD EXACT DIM. BEFORE WELDING END PLUG.

Figure 3.7

Saxton Reactor Plant
 3x3 Removable Fuel Assembly
 Fuel Rods

Assembly No.
503-4-25

Z					W
	#753 S.S. (N.R.)	#754 S.S. (R)	#755 S.S. (N.R.)		
	#759 S.S. (R)	#756 S.S. (N.R.)	#760 S.S. (R)		
	#757 S.S. (N.R.)	#761 S.S. (R)	#762 S.S. (N.R.)		
Y					X

ORIENTATION OF THE MODIFIED SAXTON 3x3 ASSEMBLY WITH IMPROVED GRIDS

Figure 3.8

Legend: S.S. - Stainless Steel Clad
N.R. - Non-Removable
R - Removable

TABLE 3.2

FUEL ROD SUMMARY OF SAXTON SPECIAL 3x3
ASSEMBLY WITH IMPROVED GRIDS

Assembly Identity	Saxton Core Position	Fuel Rod No.	Cladding Material	Fuel Rod Type	No. of Fuel Pellets	Enrichment (% U235)	UO2 (gm)	U235 (gm)
503-4-25	N-1	753 ⁽¹⁾	304 S/S	Non-Removable	50	5.71	602	30
503-4-25	N-1	754 ⁽²⁾	304 S/S	Removable	50	5.71	600	30
503-4-25	N-1	755	304 S/S	Non-Removable	50	5.71	603	31
503-4-25	N-1	756	304 S/S	Non-Removable	50	5.71	602	30
503-4-25	N-1	757 ⁽³⁾	304 S/S	Non-Removable	50	5.71	602	30
503-4-25	N-1	759	304 S/S	Removable	50	5.71	603	31
503-4-25	N-1	760	304 S/S	Removable	50	5.71	602	30
503-4-25	N-1	761	304 S/S	Removable	50	5.71	602	30
503-4-25	N-1	762	304 S/S	Non-Removable	50	5.71	602	30

- Notes: (1) Fuel rod was assembled in argon gas containing 50 ppm H₂O
 (2) Clad was heat treated to produce controlled degree of sensitization
 (3) Same as (1) except 13 ppm H₂O

The assembly will be removed during the January 1965 shutdown. At this time, the removable rods will be transferred to the Westinghouse PIF hot cells at Waltz Mill for visual examination and crud sampling. It is planned to replace these rods with Zircaloy clad rods, one of which will have been previously exposed to approximately 13,000 MWD/MTU. The assembly will then be re-installed into a peripheral location of the core for continued exposure.

303 Development of Advanced Fuel Assembly
T. Kvamme

The analysis of the results of the two phases of the 2500 hour Reactor Evaluation/Channel test of the full size prototype rod cluster control assembly utilizing Zircaloy and stainless steel fuel rods and stainless steel grids and enclosure can are being summarized. The fretting condition observed at the points of contact between the fuel rods and grid assembly spring clips is being correlated with the physical location of the fuel rods in the fuel assembly, grid spans, rod end overhang, spring finger contact pressure loading techniques, fuel rod material and manufacturing processes and flow velocity.

303.6 Development of Advanced RCC Canless Fuel Assembly Design
H. N. Andrews, N. J. Georges, T. S. Kvamme

Prototype RCC Canless Fuel Assembly Fabrication

The fabrication of the prototype canless fuel assembly for testing in the Reactor Evaluation Channel was completed. As previously reported,¹ the grid material was changed from Type 316 stainless steel to Inconel-718, an age-hardenable alloy with tensile properties that are dependent upon the aging treatment. The change to Inconel-718 permits higher spring loading and results in less yielding of the spring than with stainless steel.

The development of special fixturing techniques and brazing procedures became necessary in order to obtain satisfactory, consistent brazes with the new grid material. The original approach involved fixturing each grid such that each cell in the grid and its associated internal spring and support dimples were positioned on an ideal lattice of 0.556 inch. An aluminum oxide pin, centerless ground to 0.3950/0.3945 inch was inserted through the unbrazed grid into a mating Inconel-600 fixture plate with holes on the same 0.556 inch pitch. Difficulties were encountered in properly positioning certain pins due to slight inaccuracies of grid pitch at some internal and corner cells. In addition, slight deviations in the some of the dimple locations used to fix the free tab of the interlocking grid straps were partially responsible for both dimples not making contact with the pins.

Brazing Development

The first approach at brazing involved the fabrication of a number of 3x3 grid assemblies for use in the Saxton Reactor. Inconel-718, as received from the supplier, was used as the grid material. The grids were cleaned in nitric-hydrofluoric acid mix, rinsed with demineralized water, air dried and fixtured as previously described. Microbraz-30 was then applied to the brazing areas. The grids were placed in a vacuum furnace at approximately 0.02/0.03 micron and brazed at 2160°F. The brazes were of such excellent quality that the grids were used in the Saxton element. A number of additional samples of the as-received Inconel-718 grid material for the prototype RCC fuel assembly were

¹

WCAP-3269-5 "Large Closed Cycle Water Reactor Research and Development Program Progress Report for the Period July 1 to September 30, 1964."

also successfully brazed. It should be noted that a laboratory-type vacuum furnace was used to braze the Saxton grids and the Inconel-718 samples for the RCC prototype.

The first series of grids for the prototype RCC fuel assembly were then fabricated. Inconel-718 stock of the correct thickness was not immediately available at that time. Oversize stock was used for the internal grid straps for the prototype assembly and ground to the correct thickness. The grinding operation resulted in a different surface finish for the as-built grids than would normally be obtained when using as-rolled stock. The grids were cleaned, fixtured with pins and Microbraz-30 was applied.

The brazing temperature of 2160°F required for Microbraz-30 proved to be detrimental to the physical properties of the grid and fixture material. Microbraz-50 was substituted as a brazing alloy to permit brazing at the more acceptable lower temperature of 1850°F. Flash nickel-plating of the grids was also incorporated at this point as a surface preparation method. Good quality brazes were obtained for sample grids but without the high clear luster of the finished assemblies associated with previous WAPD-brazed grids. Brazing in an extra-dry hydrogen atmosphere and the use of an argon pre-purge were introduced as an alternate to vacuum-brazing. Other changes included picture-frame fixturing and the re-use of self-fixturing projections in line with the strap intersect slot. The grids for the REC test assembly and the mechanical test assembly (seven for each assembly) were then brazed with good-to-excellent results. Two of the grids were brazed in a 0.5 micron vacuum and the other 12 in a hydrogen atmosphere. It was demonstrated that when the nickel plating is continuous and adheres well to the base metal, excellent brazes are obtained in either hydrogen or vacuum. If the plating is discontinuous, or does not adhere to the base metal, the brazing alloy flows erratically and produces poorly brazed joints.

Brazing procedures and nickel-plating specifications are being prepared based on the preceding results.

Welding of Fuel Assembly Skeleton and Fuel Rod Loading

Temporary fixturing methods were used to align the grids, guide thimbles and top and bottom end plates prior to welding at the Cheswick Fuel Manufacturing Facility. Slight shimming of the guide thimbles was required to compensate for minor irregularities in grid pitch. Carry-over of Microbraz-50 onto the grid tabs and burn-through of the 0.012 inch thick guide thimbles and 0.015 inch tabs during welding resulted in a number of poor welds. Improvements in welding techniques and the use of better heat sink backups should alleviate burn-through during welding. The defective welds were repaired and the fuel assembly skeleton was shipped to the WAPD Forest Hills Site for inspection and drag tests prior to fuel rod loading. Previously undetected burn-through of certain guide thimbles caused the control rods to bind in the lower 18 inches of the dashpots. A sizing mandrel was inserted in each guide thimble until free passage was obtained. Dry drag tests were performed indicating slightly higher drag in the thimble dashpot area than had been previously encountered in the two previous prototype assemblies.

At this point loading of the individual rods was begun. Prior to this, some experimentation with various lubricants to reduce loading friction had been performed. Of the various lubricants tried, plain demineralized water continuously wetting the rod and grid fingers proved to be one of the better methods.

Since frictional drag and related loading forces with the new grid design are much higher a new method for loading was devised. This consisted of a hydraulic ram or piston which pushed the fuel rod smoothly while fully supported in the cylinder into the fuel assembly. Hydraulic forces of up to 1250 psi were used with a cylinder area ratio of 3 or approximately 400 lbs. maximum loading force. No major problems were encountered with this system and it is being considered for production loading.

The top and bottom end plates were welded in place with no problems encountered. Additional dry drag tests with the control rod were conducted with similar results indicating no additional friction was induced by the loading of the rods.

Lift Test

The Thermal Hydraulics Group had indicated probable lift of the fuel assembly at approximately 100% of the flow conditions for the San Onofre Plant. A lift device was designed to establish at what channel flow the prototype actually lifted. This device along with associated flow schematic can be seen in Figure 3.9.

3.25

Figure 3.9

Fuel Assembly Lift Test Schematic and Piston Assembly

As can be seen in the schematic a ΔP across the vessel internals is used to supply a flow through a calibrated ΔP coil. This flow is controlled by a shut-off piston attached to the top fuel element assembly plate and to two cylinders at the lower end of the guide tube. As the fuel element lifts, ports in the side of the cylinder are closed off and this, in turn, cuts off flow through the 3/8 in. tube and the ΔP is lost in the measuring system. At this point the fuel element would be lifted.

The general procedure will be to increase flow until the above condition takes place and then read the flow from the main loop and each vessel venturi. Then flow will be decreased to attain maximum ΔP on the 3/8 line showing that the fuel element has dropped. Maximum motion is 5/8 inches of travel for fuel element lift.

Additional data will be taken by pumping over-pressure into the cylinders by closing the high pressure side of the external system. This will permit measurement of the life forces exerted upon the bundle at flow rates in excess of that required for lifting.

Channel Loading

The finished assembly was loaded into the channel December 21, 1964 without any difficulties. Loop operation began on December 30, 1964.

Photographs

Figure 3.10 is a photo of the new top plate design for the canless assembly. The pistons for the lift device can be seen in the upper left and lower right corners. The holes in the opposite corners are for the fuel element guide pins attached to the top core plate. The photo also shows the delicate weld of the guide tubes to the top plate assembly.

Figure 3.11 is a photo of the new bottom plate design. The instrumentation tubes can be seen along with the attachment of the thimble tubes. Again the holes in the two corners are for fuel element location pins attached to the lower core plate. On the two previous designs the pins were attached to the fuel element.

Figure 3.12 shows a close-up of the top end of the fuel assembly without the top plate in place. The grid to thimble tube weld tabs can be seen, along with mixing vanes, dimples and holding springs.

Figure 3.10
Top Plate - RCC Canless Assembly

Figure 3.11

Bottom Plate - RCC Canless Assembly

Figure 3.12

Partially Loaded RCC Canless Assembly

Figure 3.13 is a photo showing the overall fuel element again partially loaded. It is taken from the bottom end and also shows internal fuel element instrumentation tubes.

Figure 3.14 is a close-up of a finished grid assembly for use in the RCC Canless Channel Tests.

Fuel Rod Vibration Studies

The system characteristics of the fuel rod vibration test program were reviewed. A 24 inch span of fuel rod was tuned to resonance at various amplitudes. No change in resonant frequency was observed. The resonant frequency of the 24 inch span was measured to be 61 cps. The calculated frequency using the simple beam formula for natural frequency was found to be 61 cps also.

Study of amplitude versus frequency was hampered because of extraneous forcing functions due to wall vibration, wooden frame vibration of coils and microscope on their respective mounts. A heavy steel beam which is to be substituted for the wooden frame will be shock mounted to the wall and rest on a rubber pad on the floor. The microscope will be floor mounted. Coils and fuel rod grids will be rigidly mounted to the steel beam.

Figure 3.13

RCC Canless Assembly Before Loading Rods

Figure 3.14

RCC Canless Assembly Grid

308.6 Reactor Evaluation Channel Operation and Special Equipment Maintenance
J. A. Santavy F. S. Faron T. D. Stormer

All modifications were completed for the conversion of the Loop D facility to fully automated operation, together with shakedown runs to demonstrate and evaluate performance of new control design. All alarms, trips, and inherent safety features were found satisfactory in accordance with design criteria. The facility was therefore released for operation with minimum manpower coverage.

Some of the modifications to Loop D for its conversion to fully automated operation included:

- 1) Removal of the pressurizer, thereby reducing the volume of water that has to be treated for chemistry control by 400 gallons.
- 2) The relief valve safety system was replaced with one consisting of a rupture disk backed up by a relief valve. The new relief system is now located directly on the main loop piping.
- 3) Two banks of heaters were added to the loop piping to provide an additional 60 Kw of heat capacity.

In addition to the shakedown runs to check-out the control philosophy for fully automated operation, the new "Canless" fuel assembly was installed in the Reactor Evaluation Channel together with its Rod Cluster Control (RCC) type control rod. The operation of the RCC control rod was then demonstrated in the new "Canless" fuel assembly by "stepping" it over its full length of travel and scramming it several times while the loop was at 100°F and 1800 psig. The flow through the fuel was 1000 gpm during these shakedown tests.

Having demonstrated operability of the RCC control rod, the experimental program defined in Sub-task 303.6¹ was initiated to evaluate the performance, life, and hydraulic characteristics of the "Canless" fuel assembly and RCC control rod in water at 575°F and 1800 psig conditioned for chemical shim control.

¹WCAP-3269-5 "Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period July 1 to September 30, 1964".

4.00 CHEMICAL SHIM DEVELOPMENT

P. Cohen, Manager
G. R. Taylor, Manager

Chemical Development
Chemistry

401 Out-of-Pile Corrosion Testing D. D. Whyte

The two sets of Inconel-600 coupons and pipe specimens described in WCAP-3737¹ and WCAP-3269-5² were removed from Loop A for chemical defilming and examination following the completion of the three-month corrosion test under chemical shim operating conditions. Test conditions were as follows:

Loop Temperature	600°F
Loop Pressure	1800 psig
Loop Flow	150 gpm
Flow Velocity through Constant Velocity Holder	16 fps
Flow Velocity through Variable Velocity Holder	6 to 20 fps
Purification System Temperature	<140°F
Purification System Pressure	1800 psig
Purification System Flow	0.044 gpm

Loop Chemistry:

Boron	1500 ppm as H_3BO_3
Potassium	3.9 ppm as KOH
Hydrogen	30. ml (STP)/Kg H_2O
Chlorides	<0.1 ppm
Oxygen	<0.1 ppm

The first set of Inconel-600 coupons had been corrosion tested for nine months at the above conditions and then descaled whereas the second set of coupons had received two month's exposure under the same test conditions to determine the short-term behavior of Inconel.

¹A Long-Term Corrosion Test of Reactor Structural Materials in a Chemical Shim Solution, D. D. Whyte, March 1964.

²"Large Closed Cycle Water Reactor Research and Development Program Progress Report for the Period July 1 to September 30, 1964".

402 In-Pile Chemistry Testing

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A. Escape Rate Coefficient and Ion-Exchange Data

During this period the analytical phase of the fuel rod defect test program at the Saxton Nuclear Facility was begun. The purpose of this program, from the chemistry viewpoint, was to establish fission product escape rate coefficient for defects of known size and location and to obtain data on ion-exchange efficiency for fission product removal.

Figures 4.1 and 4.2 present the reactor operating history during August and September, 1964, when the defect assembly was in the reactor. Table 4.1 presents the analysis of the main coolant samples for soluble radiochemical fission products during the defect test. Included in the table is the calculated escape rate coefficient (γ_A) for I^{131} and I^{133} .^{/1} The average escape rate coefficients were 1.3 and $1.1 \times 10^{-9} \text{ sec}^{-1}$ for the I^{131} and I^{133} , respectively. These values are the averages for the last several days of operation during which time the longest steady state period was attained. The high fission product activity noted during the period following August 22, 1964, is the expected "peaking" following a reactor startup. This was not observed on August 15, 1964, since there was, in fact, no inventory of fission products in the fuel at that time.

Table 4.2 presents the main coolant radiochemical data obtained during ion exchanging for removal of the fission products. The coolant activity during ion exchanging is assumed to be described by the expression

$$\frac{d N_c}{dt} = -\lambda N_c - K_p N_c + \gamma N_f$$

where N_c = atoms of isotope in coolant

K_p = purification constant, sec^{-1}

λ = decay constant, sec^{-1}

γ = escape rate coefficient, sec^{-1}

N_f = atoms of isotope in fuel tube

^{/1} The equation and assumptions for the calculation were given in WCAP-3269-5, "Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period July 1 to September 30, 1964".

Figure 4.1 - Saxton Reactor Operating History and Chemistry

Figure 4.2 - Saxton Reactor Operating History and Chemistry

TABLE 4.1 - Saxton Primary Coolant Fission Product Activities During Defect Test With No Purification

Sample Date 1964	Sample Time	Activity in coolant dpm/cc of coolant x 10 ⁻⁵ at sampling time					Observed Atom Ratio I-131 I-133	A x 10 ⁹³ / sec ⁻¹		Comments
		I-131	I-133	I-135	Cs-134	Cs-137		I-131	I-133	
8-14	1645	N.D. ^{2/}	N.D.	N.D.	.506	2.26	-	-	-	Reactor at 20 Mw 8-15-64 at 1313 hours - Power decreased 8-18-64 at 1055 hrs.
8-15	1100	1.38	25.3	N.D.	.666	3.23	.506	.50	38.5	
8-15	1135	.379	N.D.	N.D.	.764	3.45	-	-	-	
8-15	2300	.604	5.07	N.D.	.772	3.50	1.11	.22	7.8	
8-16	1100	1.13	8.52	N.D.	.772	3.45	1.23	.41	12.9	
8-16	2300	1.08	6.52	N.D.	.598	3.30	1.54	-	-	
8-17	1100	1.17	5.00	N.D.	.562	3.23	2.17	.42	7.6	
8-17	2300	1.25	4.42	N.D.	.687	3.25	2.62	-	-	
8-18	0752	1.39	4.10	1.33	.701	3.35	3.15	.50	6.2	
8-18	1000	1.35	3.95	.951	.682	3.22	3.17	.49	6.0	
8-22	1800	13.4	-	-	.949	4.65	-	-	-	Reactor at 20 Mw 8-22-64 at 1230 hrs.
8-23	1800	12.9	-	-	.910	4.98	-	-	-	
8-24	1800	12.4	-	-	.917	4.50	-	-	-	Power increased to 23.5 Mw 8-28-64 at 1830 hrs. Reactor at 23.5 Mw for all of September.
8-25	1930	10.7	-	-	.916	4.40	-	3.9	-	
8-27	1100	6.39	1.02	2.28	.610	3.18	58.1	-	-	
8-27	1552	6.02	1.18	1.78	.610	2.82	47.3	2.2	1.8	
8-28	0825	6.56	1.29	1.00	.633	3.25	47.2	2.4	1.9	
9-8	1015	3.90	N.D.	N.D.	.537	2.55	-	1.4	-	
9-11	1045	4.31	N.D.	N.D.	.532	2.55	-	1.6	-	
9-16	1030	3.98	N.D.	N.D.	.483	2.41	-	1.5	-	
9-21	1000	2.71	N.D.	N.D.	.493	2.41	-	1.0	-	
9-23	1015	3.79	.462	N.D.	.485	2.43	76.3	1.3	.70	
9-23	1555	4.15	.668	N.D.	.482	2.50	57.6	1.6	1.0	
9-24	0845	3.85	.966	N.D.	.442	2.28	37.0	1.4	1.5	

1/ Ratio with no purification should be 38.6 at steady state.

2/ N.D. - not detected usually due to long decay time before analysis.

3/ Calculated from experimental coolant activities.

TABLE 4.2 - Saxton Primary Coolant Fission Product Activities During Defect Test With Purification

Sample Date 1964	Sample Time 3/	Activity in coolant dpm/cc of coolant at sampling time						System Volumes Through Demineralizer	Ion Exchanger 1/ Decontamination Factors (D _F)				Comments
		I-131	I-133	I-135	Na-24	Cs-134	Cs-137		I-131	Na-24	Cs-134	Cs-137	
9-24	0845	385,000	96,600	N.D.	69,500	44,200	228,000	0		2/	-	-	9-24-64 (1010 hours) started purification flow at 11.5 gpm. Reactor at 23.5 Mw.
9-24	1330	152,000	36,600	N.D.	26,700	18,500	91,600	.91	312	-	13	8.9	
9-24	1330 AX	487	N.D.	N.D.	N.D.	1,400	10,300	-	-	-	-	-	
9-24	1645	70,000	28,100	N.D.	15,900	8,310	40,300	1.8	179	-	5.0	5.1	
9-24	1645 AX	390	N.D.	N.D.	N.D.	1,650	7,900	-	-	-	-	-	
9-24	1930	31,200	14,900	N.D.	10,900	4,120	25,800	2.6	427	10.8	2.6	3.1	
9-24	1930 AX	73	N.D.	N.D.	1,180	1,610	8,240	-	-	-	-	-	
9-24	2200	19,100	12,300	11,300	1,010	2,870	14,000	3.3	277	-	2.4	1.8	
9-24	2200 AX	69	N.D.	N.D.	N.D.	1,210	7,610	-	-	-	-	-	9-24 (2210 hours) ended purification flow.
9-25	0835	38,300	50,500	19,100	33,500	2,880	13,700	-	-	-	-	-	

1/ D_F = Activity before column divided by activity after column.

2/ The decrease in Na-24 from 0845 to 1330 (after .97 system volumes of purification) indicates essentially 100% removal at the early high concentration.

3/ AX indicates after demineralizer sample.

Integrating between the limits of $t_0 = 0$, the time of starting ion exchange with fission products at a steady state concentration in the coolant, and t , we obtain

$$N_{ct} = N_{co} \left[e^{-(K_p + \lambda)t} + \frac{\lambda}{\lambda + K_p} (1 - e^{-(K_p + \lambda)t}) \right]$$

where use has been made of the fact that at steady state prior to ion exchanging $\lambda N_f = N_{co} \lambda$.

For long ion-exchange periods

$$\frac{N_{ct}}{N_{co}} = \frac{\lambda}{\lambda + K_p}$$

Figure 4.3 presents curves showing the expected decrease of I^{131} and I^{133} with time of ion exchanging as predicted by the above equation. The experimental points for these isotopes are indicated on the graph. (See key to isotopes on the graph.) Included on the figure also is the theoretical curve for non-radioactive ion exchange removal considering 100% column efficiency. It can be readily seen that, indeed, the column efficiency for the removal of the fission product iodines was essentially 100%.

Included in Table 4.2 is the radiochemical analysis of the samples for Na^{22} , Cs^{134} and Cs^{137} as well as the calculated ion exchanger decontamination factors.

Table 4.3 presents the Xe fission gas data available from the test. The escape rate coefficient calculated for Xe^{133} was $7.82 \times 10^{-9} \text{ sec}^{-1}$.

The evaluation of the escape rate coefficient for daughter products, for example Xe^{133} , may be made, provided a value for the parent (I^{133} in this case) is known or assumed. In the present experiment a value for I^{133} has been calculated.

The escape rate for Xe^{133} , therefore, was calculated from the following equation:

$$\lambda_B = \frac{\alpha \lambda_A}{\left(\frac{1}{\lambda_A} - \alpha \right)}$$

Figure 4.3 - Removal of I¹³³ and I¹³¹ Radioisotopes During Anion Demineralization

TABLE 4.3

Xe FISSION GAS DATA

Sample Designation	Fission Product Coolant Activity dpm/cc Coolant		γ Xe^{133} (sec ⁻¹)	Comments
	Xe^{133}	Xe^{135}		
9-24-64 (0845)	3.16×10^6	4.11×10^5	7.82×10^{-9}	Prior to start of demineralizer
9-24-64 (1719)	1.96×10^6	1.37×10^5	-	After 1.92 system volumes through the demineralizer
9-25-64 (0835)	2.44×10^6	1.43×10^5	-	625 minutes after purification shut off

$$\text{where } \alpha = \frac{N_B^C (\lambda_B + K_p) - N_A^C \lambda_A}{N_A^C \lambda_A (\lambda_A + K_p)}$$

N_B^C = atoms of daughter B in coolant

N_A^C = atoms of parent A in coolant

λ_A , escape rate coefficient for I^{133} , was taken as $1.5 \times 10^{-9} \text{ sec}^{-1}$ for this calculation yielding the λ_B escape rate coefficient for Xe^{133} of $7.82 \times 10^{-9} \text{ sec}^{-1}$. This is the equilibrium value assuming no purification or gas stripping.

Included in Table 4.3 is the Xe^{133} concentration after 1.92 system volumes of coolant had been circulated through the low pressure surge tank during the ion exchanging. This is the normal flow path during ion exchanging. It can be seen that the effectiveness of the LPST* as a stripper is low although the amounts of gas transfer into the LPST are significant for shielding considerations.

As a first approximation the Xe^{133} activity decreased by a factor of 1.61 during the ion exchanging process when the exponential dilution factor (i.e., assuming complete gas stripping in the surge tank) for 1.92 system volumes is about 6.8.

In summary the results show that the experimentally determined escape rate coefficients are significantly lower than the presently used design values. The escape rate coefficient of the iodine isotopes are a factor of approximately 10 less than the design values while the xenon escape rate coefficient is a factor of approximately 7 less than the design value.

Samples of the main coolant were obtained during a steady state power run after the defect test assembly was removed from the reactor to determine the "background" fission product release. Table 4.4 presents the analysis of samples obtained after four and six days at steady state power (November 9, 1964 (2100) and November 11, 1964 (1200)). The six-day values of $0.147 \times 10^5 \text{ dpm } I^{133}/\text{ml}$ can be compared to the Table 4.1 value of $1.29 \times 10^5 \text{ dpm } I^{133}/\text{ml}$ obtained on August 28, 1964 (0825), six days after 20 Mw operation. I^{133} is used for comparison since with its relatively short half-life (20 hours) it is essentially at equilibrium in several days. It can be seen, therefore, that the fission product concentration was a factor of about 9 less than it was with the defect assembly in the core. However, the I^{131} activity in the coolant was higher than anticipated after the November startup (after the defect test assembly was removed) hence, it is uncertain whether or not an unintentional defect remained in the core. Additional steady state background samples will be analyzed to verify the background release.

* Low Pressure Surge Tank

TABLE 4.4

SAXTON MAIN COOLANT SAMPLES AFTER REMOVAL OF DEFECT ASSEMBLY

Sample Date	Sample Time	Activity in Coolant (dpm/ml x 10 ⁻⁵ at Sampling Time)				Comments
		I ¹³¹	I ¹³³	Cs ¹³⁴	Cs ¹³⁷	
11-9-64	2100	1.45	0.301	.0695	.116	Reactor at power 20 MW 11-5-64 (2150) - power reduced to 10 MW on 11-11-64 (1313)
11-11-64	1200	3.28	0.147	.0709	.222	
11-12-64	0845	3.20	1.01	.0545	.191	

B. Visual Examination of the 3 x 3 Defect Assembly

The 3 x 3 defect assembly was visually examined and photographed immediately upon removal from the reactor. The general appearance of the assembly was good, showing some corrosion product deposition along the high flux areas. This assembly in comparison had somewhat more crud deposition than the 3 x 3 RCC assembly, but considerably less than the center 3 x 3 removed during the November, 1963, shutdown.

The visual appearance of the defect assembly from the standpoint of crud deposition is summarized below:

1 - area = 0 to 5" from top of core	- clean shiny oxide film, no deposit
2 - area = 5 to 15" from top of core	- light spotty deposit, somewhat more deposit on corner rods
3 - area = 15 to 21" from top of core	- uniform deposit except for area just below grid - crud had a somewhat reddish appearance as viewed through the periscope
4 - area = 21 to 25" from top of core	- fairly clean area above the grid
5 - area = 25 to 28" from top of core	- light deposit mostly on corner rods
6 - area = 28 to 40" from top of core	- clean shiny oxide film, no deposit

C. Location of the Failed Fuel Assembly

After the visual examination, the 3 x 3 defect assembly was placed in the Saxton failed fuel can, to determine the feasibility of locating defects after reactor shutdown. The location of the defect is based on the detection of the soluble daughters (iodine) of insoluble fission products (Tellurium) which escape from the fuel rod during operation and plate out on the clad surface.

The failed fuel can containing the defect assembly was allowed to stand for 10 hours. The can coolant was then recirculated for about 15 minutes prior to obtaining a sample (about 7 liters of the 28 liter can volume was recirculated). It was noted while priming the pump for recirculation, that gas bubbles rose from

the can lid, indicating that the lid was not sealed properly. Storage well samples at the time the 3 x 3 was first inserted in the can and after the 10-hour period were analyzed together with the can sample for radioactive iodines. The results of the radiochemical analysis are as follows:

Storage well at loading = $.0649 \times 10^{-3} \mu\text{c } \text{I}^{131}/\text{ml}$

Storage well at sampling = $.0825 \times 10^{-3} \mu\text{c } \text{I}^{131}/\text{ml}$

Failed fuel can = $1.13 \times 10^{-3} \mu\text{c } \text{I}^{131}/\text{ml}$

Using an average value for the storage well activity, the I^{131} activity buildup in the failed fuel can after 10 hours was about $32 \mu\text{c}$.

The "expected" I^{131} activity after 10 hours was about $38 \mu\text{c}$. This calculated value was determined:

1. using the steady state reactor power coolant activity of $4.0 \times 10^5 \text{ dpm } \text{I}^{131}/\text{ml}$ (this is an experimental value prior to shutdown).
2. assuming the same escape rate for Te^{131} and I^{131} from the fuel rod defect.
3. assuming all escaping Te^{131} plates out on the defect assembly clad surfaces.

It can be shown that 6.4 times more I^{131} is contributed to the main coolant by direct escape of I^{131} from the fuel rod than from the decay of Te^{131} plated on the exterior rod surface. The amount of Te^{131} plated on the cladding then at steady state operation can be calculated from the steady state I^{131} coolant concentration:

$$N_{\text{Te}} = \frac{\lambda_{\text{I}}}{\lambda_{\text{Te}}} N_{\text{I}}$$

N_{I} = total atoms of I^{131}

N_{Te} = total atoms of Te^{131}

λ = decay constant sec^{-1}

The amount of Te^{131} initially on the fuel rod surfaces when inserted in the failed fuel can is the steady state value corrected for the decay time (time from reactor shutdown to insertion of the assembly in the failed fuel can).

The buildup of I^{131} in the failed fuel can is predicted by the following equation:

$$N_I = \frac{\lambda_{\text{Te}}}{\lambda_I - \lambda_{\text{Te}}} N_{\text{Teo}} (e^{-\lambda_{\text{Te}} t} - e^{-\lambda_I t})$$

where t = time in failed fuel can (sec)

N_{Teo} = total atoms of Te^{131} on clad surface at time of installation in can

The above treatment also assumes no I^{131} escape from the defect proper while the assembly is in the failed fuel can. It can be shown that an I^{131} escape rate coefficient of about $1 \times 10^{-14} \text{ sec}^{-1}$ would have been required to account for the observed I^{131} buildup in the failed fuel can by direct escape from the fuel. The escape rate coefficient for I^{131} at the low temperature would be about $10^{-15} (\text{sec}^{-1})$, thus the major source of the I^{131} in the failed fuel can was external to the fuel.

The test has shown therefore that defects can be detected by isolation of the assemblies after reactor shutdown.

406 Chemical Studies of Soluble Neutron Poisons

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Saxton Crud Test Program

1. Summary and Status of the Tests

Atomic Energy Commission's approval to carry out the crud test was received, the necessary additions to the plant facilities were completed, and the crud test was initiated during this period.

The total ferrous hydroxide slurry added to the reactor system is the equivalent of about 458 grams of Fe_3O_4 . The net loss of reactivity attributed to the slurry additions was about 0.7 m ρ .* This reactivity loss was observed at 20 Mwt comparing steady state borated reactivity values before and after crud addition.

The system was borated to 478 ppm after the completion of the slurry addition. Close reactivity follow was maintained during operation at 10, 20 and 23.5 Mwt. No significant unexplained reactivity change was observed after the boron addition either without nucleate boiling (10 Mwt operation) or with nucleate boiling heat transfer from the fuel rods (20 - 23.5 Mwt operation).

A decision was made to shut down the reactor and examine the two removable 3 x 3 fuel assemblies. The center and peripheral 3 x 3 assembly were examined in the Saxton storage well. In general, the amount of deposition was small, on the average, and spotty, although several small areas contained significant deposit thickness.

2. Details of Preparation and Crud Addition

2.1 Facility Installation

An apparatus for preparing the ferrous hydroxide slurry, which is being used in the Saxton Crud Test, was designed, fabricated, and installed at the Saxton site. This apparatus is shown by Figure 4.4. A line in the Saxton Charging and Volume Control System was modified to permit injection of slurry into reactor primary coolant by using a Iapp Pulsafeeder Pump Model CP-4. This consisted of replacing an elbow with a tee and welding appropriate piping and reducers to accommodate 3/8" tubing, for

* 1 m ρ = 0.1% reactivity.

Figure 4.4 - Apparatus for Preparing Ferrous Hydroxide Slurry

connection with the pump, and also to enable the line to be capped at the conclusion of the Crud Test. Figure 4.5 shows connections between the slurry preparation apparatus and the pump as well as the tubing leading from the pump to the Charging and Volume Control System line located in the pipe tunnel directly beneath the floor of the area housing the injection apparatus. All of the apparatus has performed to design conditions with no malfunction.

2.2 Crud Addition to the Reactor System

The equivalent of 458 grams of Fe_3O_4 was added to the reactor system as $\text{Fe}(\text{OH})_2$. The crud injections were as follows:

Injection Number	Injection Date - Time	Slurry Pumping Time minutes	Reactor Power Mwt	Total Fe_3O_4 per Injection (gms)
1	11-5-64 (1330)	30	0	67
2	11-9-64 (2212)	45	20	142
3	11-10-64 (0350)	31	20	107
4	11-10-64 (0722)	45	20	142

In order to minimize loss of solids, by let-down flow, during crud injection periods the following procedure was used:

1. Drain the primary system until the water level in the pressurizer is near the low level alarm point.
2. With the let-down flow secured start make-up flow at about 3 gpm and ferrous hydroxide flow at about 675 ml/min.
3. After about one-half hour of hydroxide addition, secure the hydroxide flow and flush the inlet lines by increasing the make-up flow to about 10 gpm until the pressurizer high level alarm point is approached, at which time make-up flow is secured.
4. With the primary system isolated, follow coolant crud level until the added hydroxide has essentially all deposited on the system surfaces before repeating the addition cycle.

Figure 4.5 - Slurry Injection Pump and Associated Slurry Preparation Apparatus

Figure 4.6 presents a plot of the main coolant crud concentration during and after the first injection. The reactor power was zero and the average main coolant temperature was about 520°F.

Figure 4.7 presents a plot of the main coolant crud concentration during and after the second, third, and fourth additions. For these injections the reactor power was maintained at 20 Mwt with an average coolant temperature of about 520°F.

Neglecting the low crud concentrations prior to the start of the injection, the crud concentration at any time (t) during the injection can be defined approximately by:

$$C_T = \frac{R}{V\lambda} (1 - e^{-\lambda T})$$

R = slurry injection rate (mg/min)

V = system mass (Kg)

C_T = system crud conc. at time (T) mg/Kg

T = time (minutes) after start of injection

where

$$\lambda = \frac{.693}{t_{1/2}}$$

where

$t_{1/2}$ = crud deposition "half-life" (time required for concentration to drop to 1/2 its value)

assuming disappearance rate of the circulating solids is proportional to the solids concentration. During periods with no injection, the solids concentration should decay exponentially.

The crud deposition half-life was calculated to be between two and four minutes during the injection and about forty minutes after the injection. This may be taken as an indication that the depositing species for the reactor test was primarily ferrous hydroxide. The out-of-pile loop test, however, showed deposition half-lives of from 5 to 27 minutes indicating magnetite was the primary

4.20

Figure 4.6

Figure 4.7

depositing species. The "conversion half-life" of the slurry, that is, the time required to convert 50% of $\text{Fe}(\text{OH})_2$ to Fe_3O_4 , is 2.2 minutes. Subsequent to the time shown on Figure 4.7 the coolant crud concentration varied from 0.030 to 0.10 ppm.

Table 4.5 presents a summary of the reactor coolant chemistry during the crud test. The system chemistry was maintained essentially neutral during the crud injections. On November 13, 1964, the boron concentration was increased to about 470 ppm at 10 Mwt reactor power, for the hideout test proper. Boron was maintained in the system during the remainder of the test.

On December 27 potassium as KOH was added to the system and subsequently removed by ion exchanging to observe the pH effect on reactivity. Results of these tests are given under "Reactivity Follow".

2.3 Reactivity Follow During Crud Test

The design purpose of the crud test, as originally stated, was to establish an operational method suitable for use in a power reactor, whereby it can be shown that boron hideout is or is not occurring at a given point in time. One of the major pre-requisites for the success of "an operational method" is the ability of the physicists to predict the reactivity changes associated with controlled reactor power swings. Specifically, the reactor power swing from a normal operating level to a level sufficiently low to suppress nucleate boiling must be followed accurately. In the Saxton reactor the power swing from a nucleate boiling core condition to a non-nucleate boiling core condition would be from above 20 Mwt to 10 Mwt.

An effort was made throughout the crud test to maintain a close reactivity follow. Figure 4.8 presents a plot of the reactivity follow during November and the operating period of December, 1964. Figure 4.9 reports the coolant chemistry during these periods.

The reactivity follow (expressed on Figure 4.8 as % unexplained reactivity) is plotted in two ways:

TABLE 4.5 - Sexton Main Coolant Chemistry During Crud Test

Sample Designation			Boron	Potassium	Lithium	NH ₄ ⁺	pH at 25°C	Resistivity at 25°C x 10 ⁻³	Crud Concentration ppm
Date 1964	Time	Place	ppm	ppm	ppm	ppm	units		
11-5	0830	MC		0			7.08	580	
	1400	MC	18	< .05	< .005				2.1
11-6	1110	MC							0.2
11-9	1110	MC							
	1745	MC		.5	< .05		9.4		
	2100	MC	3.96	.09		.82			0.2
11-10	0130	MC	4.42			.76	7.8		0.6
	1030	MC	5.04				7.4		0.4
	2255	MC	4.88	.17			7.72	298	107
11-12	0845	MC	10.0				7.62		0.070
	1100	MC							
11-13	0955	MC	10.0	.17	0		8.2	280	0.050
	1100	MC							
	1550	MC	5.2				7.3	305	
	1700	MC	406				6.44	205	0.090
	1800	MC	488				6.15	220	0.090
	2000	MC	475						
	2345	MC	474	.07	0				0.070
11-14	1010	MC	476	0.1	.005		6.7		0.070
	1035	P	453	0.1	.002				
	2015	MC	472	0.1	.006		6.20	246	
11-15	0430	MC	472	0.1	.007		6.22	245	
	1145	MC	473	0.1	.013		6.30	234	0.050
	2100	MC	472	0.1	.015		6.30	238	
11-16	0345	MC	472	0.1	.010		6.25	237	
	1430	MC	472	0	.021		6.30	227	0.050
11-17	1135	MC	470	0.1	.020		6.22	216	0.040
11-18	0900	MC	470	0.1	.030		6.30	213	
	0900	P	470	0.1	.024		6.10	253	
	1430	MC	448	0.1	.038		6.20	248	0.030
	2100	MC	451	0.11	.026		6.32	223	0.066
11-19	1415	P	450	0.06	.025		6.10	232	
	1425	MC	451	0.11	.032		6.22	215	
11-20	1000	P							
	1010	MC	451		.06		6.35	213	0.066
	1450	MC	448	0.1	.06		6.12	220	0.066
11-21	0835	MC	448	.05	.062		6.13	228	0.066
12-18	0930	MC	450	.08	.036	.50	6.02		0.265
	1000	P	468	.08	.036	.25	6.09		
	1430	MC	437	.09	.027	.50	5.60		0.476
	1615	MC							0.080
	1930	MC	437	.09	.034	.37	5.80		
12-19	0830	MC							0.132
	0915	MC	438	.09	.034	.37	5.95		
12-20	1130	MC	439	.09	.038	.37	5.90		0.106
12-21	1000	MC	438	.09	.040	.37	6.05		
	1345	MC							0.080
	1415	MC							
	1455	MC							0.106
	1655	MC	439	.09	.043		5.90		0.054

(continued)

TABLE 4.5 - Saxton Main Coolant Chemistry During Crud Test (continued)

Sample Designation			Boron	Potassium	Lithium	NH ₄ ⁺	pH at 25°C units	Resistivity at 25°C x 10 ⁻³	Crud Concentration ppm
Date 1964	Time	Place	ppm	ppm	ppm	ppm			
12-22	0945	MC	440	< .05	.047		5.95	Not Measured	0.054
	1745	MC		.54					0.080
	2240	MC	435	.55	.051		6.28		0.080
12-23	0245	MC	437	.60	.049		6.25		0.106
	0950	MC	437	.55	.055		6.22		0.080
	1455	MC							0.182
12-24	1400	MC	391	.50	.063		6.18		
12-25	1300	MC	391	.50	.071		6.30		
12-27	1300	MC	392	.49	.089		6.40		
12-28	1045	MC	392	.46	.091		6.45		0.050
	1450	MC							0.053
	1740	MC	391	4.69	.096		6.90		0.061
	1900	MC		4.79					0.035
12-29	0830	MC	391	4.71	.103		6.95		0.035
	1500	MC		8.45	.103		7.18		0.035
	1630	MC	390	8.73	.103		7.15		0.044
	1735	MC	389	8.60	.101		7.22		
	2100	MC	389	8.61	.102		7.20		0.053
12-30	0950	MC	389	8.56	.104		7.30		0.044
	1530	MC	391	3.63					
	1920	MC		.54					
	2200	MC		.27					
	2300	MC		.15					
12-31	0845	MC	391	.18	0.010				0.035

4.25

UNEXPLAINED REACTIVITY DURING SAXTON CRUD TEST

Figure 4.8

SAXTON COOLANT CHEMISTRY DURING CRUD TEST

Figure 4.9

1. (X marks) the reactivity follow neglecting pH effect on reactivity,
2. (+ marks) the reactivity follow correcting for the pH effect on reactivity to a constant pH value of 6.76 (an arbitrarily chosen point).

Unexplained reactivity is defined as the difference between the observed and the predicted core reactivity at a given time during reactor operation.

The pH effect on reactivity is as yet not certainly explained; however, the effect, which briefly, shows a loss of core reactivity at low pH reactor operation and a gain at high pH operation, has been well documented. It is believed that the reactivity change, during a coolant pH change, is caused by either an in situ morphological change of the crud deposits or a relocation of the deposits. The relocation of the deposits may be from the boiling to the non-boiling surfaces, for example, or vice versa. Both the structural change and the relocation mechanisms may ultimately change the fuel temperature and/or the degree of nucleate boiling and the associated void fraction.

The correction for pH as used on Figure 4.8 incorporates the newly derived functional relationship between reactivity change and pH. This relationship, discussed below, considers the change in the pH reactivity coefficient with change in pH and replaces the previously used constant coefficient of 1.6 m ρ per pH unit (1 m ρ = 0.1% reactivity).

Several observations can be made from Figure 4.8:

1. November 1964 Period
 - a. Prior to the crud test the pH corrected unexplained reactivity loss was about 1.25 m ρ at 20 Mwt. This loss was associated with the physics follow during the boron and potassium removal prior to the crud injection.
 - b. During the crud injection and the following day the unexplained reactivity loss increased to 4.1 m ρ at 20 Mwt. (See November 9-10).
 - c. This loss was recovered and during the next steady state operation at 20 Mwt the unexplained loss was about 0.85 m ρ . (See November 18).
 - d. From a physics standpoint one of the best comparisons of the reactivity change for the crud addition can be made from the steady state (20 Mwt) borated

system prior to the crud injection and the steady state (20 Mwt) borated system after the crud injection. The unexplained reactivity losses for the condition are:

0.25 before crud (See Nov. 2.)
0.80 after crud (See Nov. 18.)

Therefore the maximum net loss attributable to the crud is about $0.6 \text{ m } \rho$. The recovery of most of the reactivity lost during the injection may have been due to either a redistribution of the crud on the core surfaces or an "aging" effect which changed the heat transfer properties of the deposit.

- e. During each of the other periods the transients caused by reactor operation were too severe to allow a good reactivity follow. Testing for the next quarter will include efforts to provide better predictions of the reactivity behavior during power swings since, as was pointed out earlier, the entire "operational method" is based on the ability to follow reactivity at different power levels.

2. December 1964 Period

- a. The steady state reactivity loss was about $2.7 \text{ m } \rho$, which is a factor of about 3 greater than the loss for the November steady state. It is believed that most of the loss displacement was due to the inability of the computer code to handle the xenon transients properly. The values reported in Figure 4.7 for December were partially hand-corrected for this error; however, it is believed that a larger correction would be valid.
- b. The new functional pH coefficient of reactivity handles the reactivity follow during coolant chemistry changes much better than the previously used constant coefficient. The corrected reactivity swing was $0.3 \text{ m } \rho$ during the Dec. 27-Jan. 1 pH transients. Whereas the use of the constant pH coefficient of reactivity would have resulted in a greater unexplained reactivity swing.

2.4 Heat Transfer Considerations

The crud pattern on the central 3 x 3 assembly as examined during the November 1964 shutdown indicates the effect of the amount or the degree of nucleate boiling on the deposition. Deposition occurred essentially only on these surfaces of the fuel rods facing the enclosure can in the high flux region. The center rod in the 3 x 3 had little or no deposition.

A possible explanation of the crud pattern may be obtained from a consideration of the assembly design. Figure 4.10 presents a sketch of the 3 x 3 center assembly and adjacent fuel rods in the 9 x 9 indicating the various flow channels for thermal and hydraulic calculations. Five basic unit cells can be defined, three in the 3 x 3 enclosure can and two in the 9 x 9 proper.

In a flowing system with an equal pressure gradient across all the parallel channels,

$$h \propto D_e^{1/3}$$

where

h = forced convection heat transfer coefficient

D_e = equivalent diameter of the flow channel

Table 4.6 presents a summary of the ratios of the convective heat transfer coefficients for the flow channels as defined on Figure 4.10. Included in Table 4.6 is the relative power generation coefficients for the different regions in the 9 x 9. These power generation coefficients are the products of a normalized flux distribution in a region of the assembly and a normalized macroscopic fission cross-section for the specific rods in that region.

The degree of nucleate boiling on the surface of a fuel rod in any channel can be determined from a reference set of conditions. The degree of nucleate boiling (q''_{NB}) can be defined

$$q''_{NB} = q''_T - h [T_w - T_b]$$

FLOW CHANNEL EQUIVALENT DIAMETER

1	.227 INCHES
2	.318 INCHES
3	.422 INCHES
4	.551 INCHES
5	.704 INCHES

FLOW CHANNELS IN CENTRAL 3 x 3 AND 9 x 9 FUEL ASSEMBLIES

Figure 4.10

TABLE 4.6
THERMAL AND HYDRAULIC RATIOS FOR THE
VARIOUS ASSEMBLY FLOW CHANNELS

Flow <u>/d</u> Channel Comparison	Convective <u>/a</u> Heat Transfer Coefficient Ratio	Power <u>/b</u> Generation Coefficient Ratio
3/1	1.231	1.162
3/2	1.095	1.162
3/4	.917	1.197
4/1	1.341	0.971
4/2	1.201	0.971
5/2 <u>/c</u>	1.305	1.448
5/4 <u>/c</u>	1.085	1.492
5/1 <u>/c</u>	1.458	1.448

/a $h_i/h_j = [De_i/De_j]^{1/3}$

/b Coefficient uses maximum spiked rod values (16 Kw/ft) for flow channels 3 and 5.

/c Using maximum fuel rod power in region just outside 3 x 3. This is highest power condition in the 9 x 9.

/d Channels indicated on Figure 4.10.

where

$$T_w = T_{sat} + \Delta T_{JL}$$

$$q_T'' = \text{total heat flux Btu/hr-ft}^2$$

$$T_w = \text{clad surface temperature } ^\circ\text{F}$$

$$T_b = \text{bulk coolant temperature } ^\circ\text{F}$$

$$T_{sat} = \text{saturation temperature } ^\circ\text{F}$$

$$\Delta T_{JL} = \text{Jens and Lottes correction for inception of nucleate boiling } ^\circ\text{F}$$

Using as a reference condition, the maximum spiked (nominal 16 Kw/ft) fuel rod in the 9 x 9 assembly, we can calculate the relative degree of nucleate boiling in each of the flow channels.

The basic parameters in the following calculations are:

1. q_T'' (spiked) = 5.38×10^5 Btu/hr-ft² (maximum flux at 20 Mwt reactor power rodded operation)
2. h (channel 5) = 2550 Btu/hr-ft² °F
3. T_b = 550°F in all channels at maximum axial flux location

A - For spiked rod in 9 x 9

$$\begin{aligned} q_{NB}'' &= 538,000 - 2550 (92) \\ &= 303,000 \text{ Btu/hr-ft}^2 \end{aligned}$$

B - For corner rod in 3 x 3

$$\begin{aligned} q_{NB}'' &= \frac{538,000}{1.448} - \frac{2550}{1.458} (92) = 371,500 - 161,000 \\ &= 210,000 \text{ Btu/hr-ft}^2 \end{aligned}$$

C - For center rod in 3 x 3

$$q''_{NB} = \frac{538,000}{1.492} - \frac{2550}{1.085} (92) = 361,000 - 216,000 \\ = 145,000 \text{ Btu/hr-ft}^2$$

D - For side 3 x 3 rod

$$q''_{NB} = \frac{538,000}{1.448} - \frac{2550}{1.305} (92) = 372,000 - 180,000 \\ = 192,000 \text{ Btu/hr-ft}^2$$

NOTE: The calculated nucleate boiling values given above are relative to the spiked rod presumed operating at 16 Kw/ft. This is, however, believed to be much higher than the actual power value.

It can be seen, therefore, that the amount of nucleate boiling on the spiked rods is expected to be higher than on the non-spiked rods. Also the reason for crud deposition on the 3 x 3 rods facing the enclosure can (channels 1 and 2) as compared to the center rod (channel 4) and the surface of the rods facing the center rod may be explained by the significant difference in boiling in these channels. During the Loop B tests significant difference in deposition was observed in tests 4 and 5.^{/1} Test 4 operated at from 88,000 to 126,000 Btu/hr-ft² of calculated nucleate boiling while test 5 operated from 159,000 to 173,000 Btu/hr-ft² of boiling. Test 4 showed about 35 mg crud/dm² while test 5 showed about 200 mg/dm² on the heaters.

The effects of crud deposits on these calculations cannot be evaluated quantitatively, however, since the deposit roughness effect on heat transfer, for example, has not been thoroughly defined.

^{/1} WCAP-3269-3, "Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period April 1 to June 30, 1964", p. 4.30.

Although the degree of nucleate boiling is important in crud deposition, other factors must be considered. The coolant chemistry and the operating time can also effect the amount of deposition. The 3 x 3 fuel assembly (503-4-21) which was removed from the center core position during the November 1963 shutdown showed considerably more deposit in the high flux region, over each of the 9 fuel rods. This assembly operated at essentially the same power generation rate as the crud test 3 x 3 assembly. The operating differences between the two assemblies were basically operating time and chemistry:

Crud test assembly: ~ 1.5 month operation at
power, ~ 450 ppm B,
~ 0.1 ppm K, .005 to .071
ppm Li

November, 1963 assembly: ~ 7 months operation
at power, 1200 to 750
ppm B, ~ 3.0 ppm K,
.1 to .4 ppm Li

A review of all the fuel rod examinations will be performed in an effort to evaluate the "time and chemistry effects".

3. Visual Examination of Crud Test 3 x 3 Fuel Assembly

During the reactor shutdown of November, 1964, the center and the peripheral 3 x 3 fuel assemblies were examined at the Saxton site. The purpose of the examination was to determine the extent of crud deposition from the $\text{Fe}(\text{OH})_2$ injections.

1. Peripheral Location 3 x 3 Fuel Assembly:

This assembly showed no deposition on any of the fuel rod or enclosure can surfaces. The surfaces had the typical shiny oxide appearance.

2. Center Location 3 x 3 Fuel Assembly:

Deposition was observed along sections of this assembly between 15 and 23 inches from the tops of the fuel rods. This is the region of maximum flux (between the second and third grids from the top). The deposit appearance was black in contrast to the somewhat reddish-black appearance of other 3 x 3 fuel assembly deposits previously examined at Saxton.

Generally, the crud deposition occurred only on those portions of the fuel rod circumferences that faced the enclosure can walls. That is to say, the corner rods had crud around about 180° of the circumference, while the side-center rods had deposits on about 90° of the circumference. The center fuel rod did not appear to have any deposit. This rod was viewed in the assembly, while one side-center rod was removed. The side-center rods #754 and #761 showed the before mentioned pattern of deposition only on the enclosure can wall side. Fuel rod #759 showed some deposition on its inside surface.

Although several spotty areas showed fairly significant thickness of crud, the general deposit appeared to be less than that observed on the 3 x 3 fuel assembly removed from the reactor during the November, 1963, shutdown. That deposit was determined in the W PIF hot cells to be between 60 and 70 mg per dm^2 . It is estimated that the average deposit surface concentration on the crud test 3 x 3 fuel rods was about 10 to 30 mg/dm^2 on the deposited areas. The enclosure can showed essentially no deposition.

An unsuccessful effort was made to obtain a crud sample. Three side-center rods were removed from the 3 x 3. Sampling was attempted on two of these. These efforts were unsuccessful due to the small amount of deposit in the first place and secondly, due to the difficulty of underwater sampling. The fuel assembly outer surfaces were then brushed with a nylon brush mounted on a magnet. Again, the effort to collect a sample failed although the deposit was almost entirely removed from the surfaces.

Figure 4.11 shows the fuel rod orientation in the crud test 3 x 3 fuel assembly.

Figures 4.12, 4.13, and 4.14 show pictures of the 3 x 3 assembly as viewed in the Saxton storage well.

W			X
	755	760	762
	754	756	761
	753	759	757
Z			Y

Figure 4.11

FUEL ROD ORIENTATION OF THE CRUD TEST 3 x 3 FUEL ASSEMBLY

(All Stainless Steel Rods)

View in area half-way
between second and
third grid with side
center rod removed.

Figure 4.12 - Saxton Crud Test 3 x 3 Fuel Assembly (X - Y Face)

View in area half-way
between second and
third grid from the
top of the core.

View just above third
grid from the top of
the core.

Figure 4.13 - Saxton Crud Test 3 x 3 Fuel Assembly (W - Z Face)

View in area just below
first grid from the top
of the core.

View in area just below
second grid from the top
of the core.

Figure 4.14 - Saxton Crud Test 3 x 3 Fuel Assembly (W - Z Face)

407 Chemical Shim Kinetic Experiments
V. Rajagopal, G. E. Swen

Several oscillation and noise measurements were performed at Saxton to study the effects of synthetic crud addition and pH effects on reactor operation with and without boric acid in the main coolant. These tests were performed to study the effects on reactor parameters such as Doppler reactivity, fuel time constant and nucleate boiling.

Oscillation tests consist of modulating reactivity through a reactivity modulator at various frequencies and measuring the reactor flux and temperature responses by use of cross-correlation techniques. Doppler coefficient of reactivity and fuel time constant can be determined from this data. Doppler coefficient of reactivity is defined as the reactivity change associated with fuel temperature change per unit change of power at the operating conditions ($\alpha_f \frac{\partial T_f}{\partial P}$). Fuel time constant is defined as the time it takes for the fuel temperature to reach 63.2 percent of its final value for a step change in nuclear power.

Crud Test

Crud Injection and Reactivity Effects

Saxton synthetic crud injection test commenced on November 9th. The plant was operating at a power level of 20MWt without any boron in the main coolant. Synthetic crud injection started on November 9th. Any reactivity change due to crud addition was compensated by coolant temperature change and rod motion.

Plots of the chemistry and reactivity follow data during crud injection are shown in Figure 4.15. Details of these follows are reported under LRD-406. The reactivity shown in Figure 4.15(b) is unexplained reactivity that can be attributed to crud addition.

Nucleate Boiling Measurements

Previous noise measurements at Saxton have shown that nucleate boiling in a reactor core operating at 2000 psi and 530°F T^{ave} can be measured as a resonance in the power spectrum of ion chamber fluctuations between 10 and 20 cycles per second (TTP-7679¹). Many of the recent measurements experimentally confirm the relationship between the resonance and nucleate boiling. For example, at 10MWt at Saxton a small resonance appeared at normal operating conditions. But it completely disappeared when the pressure was increased and average coolant

¹Reactor Noise Measurements on the Saxton Reactor, V. Rajagopal, June 1964 (AEC Symposium Series No. 4 - Noise Analysis in Reactor Systems).

Figure 4.15 SAXTON CRUD TEST - Reactivity Effects During Crud Injection
20MWt - No Boron

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Subsequent oscillator measurements show that the increased Doppler reactivity coefficient gradually returned to the value prior to crud addition. Detailed analysis of several oscillator test data are continuing and will be reported later.

pH Tests

pH Effect on Reactivity

A potassium addition and removal experiment was conducted at Saxton between December 27th and January 1st.

The plant was operating at a power level of 20MWt with a boron concentration of about 330 ppm in the main coolant. Addition of potassium started on December 28th. Any reactivity increase in the core was compensated by letting the coolant average temperature increase.

A summary plot of reactivity and chemistry follow data is shown in Figure 4.17. Details are reported under LRD-406. The reactivity shown in Figure 4.17 is unexplained reactivity that can be attributed to pH effects.

Nucleate Boiling Measurements

Data obtained from the Saxton reactor during the test are shown in Figure 4.18. As mentioned before, the resonance at 12 cps is believed to represent nucleate boiling in hotter regions of the core and the one at 16 cps nucleate boiling in more general regions.

This data shows that nucleate boiling in hotter regions of the core increase at high pH while that in general regions decrease.

Oscillator Measurements

The results obtained during the pH test are summarized below.

Parameter	20MW Boron Low pH (60)	20MW Boron High pH (7.4)
Doppler Coefficient $10^{-3}/\text{MW}$	0.310	0.292
Fuel Time Constant sec.	2.6	2.2

Figure 4.17 SAXTON pH TEST - Reactivity Effects
20Mwt

Figure 4.18 SAXTON pH TEST - Nucleate Boiling Effects

20Mwt
530°F
2000 PSI

Similarities between crud injection and pH change effects on nucleate boiling, Doppler coefficient of reactivity and fuel time constant are considered more than coincidental. Further analysis of data to explore the significances of these similarities will be carried out and reported later.

5.00 PLANT DEVELOPMENT

J. M. Gallagher, Manager
P. B. Haga, Manager

Systems Analysis Section
Systems Engineering Section

501 System Stability and Control Studies
A. Batenburg, G. R. Buis

The results of the automatic boron control study described in WCAP-3269-5¹ have been summarized for incorporation in WCAP-3269-45 "Reactor Control and Protection System Functional Requirements". The report will also include a description of the parametric study of the temperature, power and pressure constraints maintained by the reactor control and protection system and the sensitivity of operating margins as a function of design parameters.

503 Development of Auxiliary Systems
H. J. Cordle

The preparation was begun of WCAP-3269-31 "Development of Auxiliary Systems" which will summarize the results of the development of the various auxiliary system for closed-cycle water reactor with chemical shim control.

¹Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period July 1 to September 31, 1964.

6.00 REACTOR HAZARDS ANALYSIS

R. A. Wieseemann, Manager Plant Safeguards & Licensing

Safeguards Analysis

J. D. McAdoo

A summary outline was prepared for a systematic safety analysis of the closed cycle water reactor and associated plant features. When completed, this analysis will serve as a topical report covering various aspects of the effort under this task. Work accomplished to date includes formulation of safety principles and objectives for the core, reactor systems, control and instrumentation. As the status of this work is preliminary, a more detailed report is being deferred until the next quarter.

7.00 CRITICAL EXPERIMENTS

G. H. Minton, Manager
D. F. Hanlen, Manager

Reactor Physics & Mathematics Section
Reactor Evaluation Center

701 Critical Experiment Planning and Analysis

H. T. Williams, W. J. Eich

The report titled, "The Analytic Representation of Rod Cluster Control Elements in Nuclear Design Calculations," (WCAP-3269-38) was completed and submitted to the AEC for approval. Final effort on the report involved an extension of the comparison between analysis and experiment to include the effective multiplication factor for rodded and unrodded core configurations. Previous correlation had been in terms of integral rod worth measurements obtained by use of the dp/dH curve for the reactor. For the recommended analytical procedure, the calculation predicted the experimental results to an accuracy of $\pm 3\%$ on the basis of either of the two correlations mentioned above.

702 Performance of Critical Experiments

R. D. Leamer, G. N. Hamilton, D. F. Hanlen, J. P. Tobin

A. RCC Experiments in 2.72% Zr-4 Clad 3.05:1 W/U Cores

Reactivity worths of uniform arrays and clusters of Ag-In-Cd were measured in square cores of 2.72% fuel with zirconium-4 cladding in the 0.600-inch lattice.

Table 7.1 represents the data from the clean lattice measurements. Table 7.2 presents the data from the uniform array experiments, and Table 7.3 presents the data obtained from the cluster experiments. Corresponding core diagrams are included in Figures 7.1 thru 7.7.

In Figure 7.8 the inverse cube root of ρ/dh is plotted against the corrected water height to indicate the reliability of the data used for reactivity calculations. Reactivity worths were calculated from

$$\rho \left| \begin{array}{c} H_2 \\ H_1 \end{array} \right. = \frac{21.74}{(0.23H + 0.30)^2}$$

and are in dollars. Since the temperature of the core remained practically constant throughout the experiments, no reactivity corrections were made.

TABLE 7.1

Clean Lattice Experiments:
3.05 W/M Ratio, .600" Pitch, 2.72% Enriched UO₂

<u>Item</u>	<u>Configuration</u>	<u>Fuel Loading (rods)</u>	<u>Figure No.</u>	<u>Critical Water Height (cm)</u>
1	Best Circle	491	7.1	139
2	26 x 26 rectangle	676	7.2	55.75
3	27 x 27 "	729	7.3	50.51
4	28 x 28 "	784	7.4	46.46
5	31 x 31 "	961	7.5	38.58

TABLE 7.2

Uniform Array Experiments
3.05 W/M Ratio, .600" Pitch, 2.72% Enriched UO₂ Zirc Clad Fuel

Item	Configuration	Fuel Loading (rods)	Fig. No.	Critical Water Height (cm)	$\Delta k/k - \beta$	
					Ref. to Clean Core	Ref. to Water Holes
1	25-.403" Ag-In-Cd Rods	936	7.5	92.15	11.71	12.43
2	25-.403" Ag-In-Cd Rods w/.408" ID-.010" wall SS cladding	936	7.5	93.82	11.82	12.55
3	25 Water Slots	936	7.5	37.41	- 0.72	0
4	25-.260" Ag-In-Cd Rods	936	7.5	57.13	7.07	7.80
5	25-.330" Ag-In-Cd Rods w/.415" OD-.040" wall SS cladding	936	7.5	70.53	9.54	10.26
6	25-.330" Ag-In-Cd Rods	936	7.5	69.44	9.38	10.10
7	25-.330" Ag-In-Cd Rods w/.335"-.020" wall SS cladding	936	7.5	70.03	9.47	10.19
8	25-.395" ID-.020" wall SS thimbles-flooded	936	7.5	38.41	- 0.10	0.62
9	25-.415" OD-.040" wall SS thimbles-flooded	936	7.5	39.19	0.36	1.08
10	16-.403" Ag-In-Cd Rods	768	7.4	102.15	8.51	9.05
11	16-.330" Ag-In-Cd Rods	768	7.4	79.91	6.85	7.39
12	16-Water Slots	768	7.4	45.13	- 0.54	0
13	9-.403" Ag-In-Cd Rods	720	7.3	76.51	5.05	5.40
14	9-Water Slots	720	7.3	49.44	- 0.35	0

TABLE 7.3

Cluster Experiments

3.05 W/M Ratio, .600" Pitch, 2.72% Enriched UO₂ Zirc Clad Fuel

Item	Configuration	Fuel Loading (rods)	Fig. No. 7.6	Critical Water Height (cm)	$\Delta k/k$ - \$ Ref to Clean core	Ref to Water Holes
1	16-.403" Ag-In-Cd Rods	884	A	100%	Subcritical	
2	16-.330" Ag-In-Cd Rods	884	A	89.70	*	11.20
3	16-.330" Ag-In-Cd Rods w/.415" OD-.040" wall SS cladding	884	A	91.86	*	11.36
4	16 Water Slots	884	A	39.14	*	0
5	16-.415" OD-.040" wall SS thimbles-flooded	884	A	42.27	*	1.66
6	16-.403" Ag-In-Cd Rods	945	A	89.75	11.53	12.38
7	16-.330" Ag-In-Cd Rods w/.415" OD-.040" wall SS cladding	945	A	73.29	9.91	10.76
8	16 Water Slots	945	A	37.21	- 0.85	0
9	16-.415" OD-.040" wall SS thimbles-flooded	945	A	39.56	0.57	1.42
10	16-330" Ag-In-Cd Rods	945	A	71.75	9.70	10.56
11	16-Water Slots	945	B	37.17	- 0.88	0
12	16-.403 Ag-In-Cd Rods	945	B	91.12	11.63	12.51
13	16 Water Slots	945	C	37.28	- 0.81	0
14	16-.403" Ag-In-Cd Rods	945	C	75.81	10.22	11.02
15	16 Water Slots	945	D	37.24	- 0.83	0
16	16-.403" Ag-In-Cd Rods	945	D	94.75	10.84	12.72
17	12 Water Slots	949	E	37.51	- 0.66	0
18	12-.403" Ag-In-Cd Rods	949	E	67.63	9.11	9.77
19	20-.330" Ag-In-Cd Rods	941	F	94.28	11.85	-
20	20-.403" Ag-In-Cd Rods	941	F	Subcritical	-	-

* Clean core data not established.

FIGURE 7.1

491, 2.72% Zr-Clad Fuel Rod
Loading in 0.600-inch Lattice.

FIGURE 7.2

676, 2.72% Zr-Clad Fuel
Rod Loading in 0.600-inch
Lattice.

FIGURE 7.3

720, 2.72% Zr-Clad Fuel Rod
Loading with 9 Test Positions
in 0.600-inch Lattice

FIGURE 7.4

768, 2.72% Zr-Clad Fuel Rod
Loading with 16 Test Positions
in 0.600-Inch Lattice

FIGURE 7.5

936, 2.72% Zr-Clad Fuel Rod
Loading with 25 Test Positions
in 0.600-Inch Lattice

○ FUEL RODS
● AG-IN-CD RODS

CLUSTERS USED IN 30 x 30 AND 31 x 31 ROD CORE LOADINGS

Figure 7.6

FIGURE 7.7

30 x 30 Core Loading with 884, 2.72%
Zr-Clad Fuel Rods and 16 Test Positions
in the 0.600-Inch Lattice. A Typical
Cluster is Shown.

B. Meniscus Effects in Partial Height Reactivity Measurements

In an attempt to minimize errors associated with reactivity measurements, a series of three experiments has been performed to determine the effects of meniscus on differential moderator worth. Each experiment was performed in two ways: (1) measuring reactivity as a function of decreasing moderator level and (2) measuring reactivity as a function of increasing moderator level and comparing the resultant just critical moderator levels, Figures 7.9, 7.10, and 7.11.

The experimental procedure consisted in overfilling the moderator system to such an extent that draining was necessary to control excess reactivity as control rods were withdrawn to their upper limit, thus insuring that a positive (or concave) meniscus was formed between the fuel and moderator surface. The moderator was lowered in steps recording both moderator level and reactivity. This procedure was continued until the reactor was subcritical by approximately ten cents ($k_{eff} \sim 0.9993$) with the control rods fully removed. At this point the moderator level was lowered an additional one centimeter and filling operations commenced, resulting in an inversion of the meniscus (now convex or negative). The same data were recorded as previously described until the reactor was approximately twenty cents supercritical ($k_{eff} \sim 1.0015$) at which point the experiment was terminated. It should be pointed out that all measurements were made at least two decades above source level, thereby insuring maximum instrumentation response, even though subcritical at certain points.

A plot was made of the inverse cube root of the reactivity difference associated with the change in meniscus and as expected the curve extrapolated to the total axial reflector savings (~ 13.0 cm) indicating that the error is directly proportional to differential moderator worth, Fig. 7.12. It is felt that although the error exists, no correction is necessary to experimental data since all reactivity measurements are made in one

FIGURE 7.9
Reactivity versus Water Height
Critical Height 41.0

FIGURE 7.10
Reactivity versus Water Height
Critical Height 50.0

direction, e.g., reactivity decreasing with decreasing moderator level. Further, the error associated with meniscus effects is within the experimental uncertainties associated with moderator level measurements. Table 7.4 lists moderator levels and reactivity errors associated with meniscus inversions.

TABLE 7.4

Moderator Levels and Reactivity Errors
for Meniscus Inversions

<u>Moderator Heights (cm)</u>	<u>Reactivity (ρ)</u>
40.98	5.20
50.05	3.45
77.72	1.20

C. Critical Size

The critical size of the 2.72% Zr-4 clad fuel in the 0.600 inch lattice has been measured. The smallest core operated is shown in Figure 7.1, which had k -excess = 0.078 \$ at 17°C, fully flooded. With a peripheral fuel rod worth of 4.6¢ per rod, the just critical fuel rod loading is 489.3 ± 0.2 .

D. Buckling and Reflector Savings

Buckling and reflector savings have been measured for three clean cores of 2.72% Zr-4 clad fuel in the 0.600-inch lattice plates. The best value for the just critical buckling is $86.4 \pm 1.1 \text{ M}^{-2}$.

Buckling and reflector savings values have been obtained from fuel rod scans in the 23 x 23, 24 x 24, and 26 x 26 rod cores. In all cores the horizontal power shape was measured using one line of fuel rods thru core center, and counting them at four or five different heights. The vertical power shape was obtained by counting every other rod at one centimeter intervals along its length. The data from several comparable scans in the same run were averaged and normalized to unity. The arc-sine of the normalized power was then plotted against core position; Figures 7.13 and 7.15 are typical vertical plots and Figure 7.14 is a typical horizontal plot. A straight line drawn thru these points gives buckling and reflector savings values. All scans were hand fit in this way and at least one scan from each run was analyzed with COFIT. In all cases the machine fits agreed well with the hand fits if the aluminum spacer plate was not important (Figures 7.13 and 7.14).

Figure 7.15 shows the rectified power shape for a case with the aluminum plate. It is apparent that one straight line cannot fit the points above and below the plate, which COFIT must attempt. The solid line segments are the ends of the COFIT calculation. The dashed line segments are the ends of the two separate lines which appear to be better fits above and below the plate. The top and bottom reflector savings values from the hand fits agree well with the results from larger cores (e.g., Figure 7.13), but the COFIT values do not.

However, the total reflector savings from the hand fits agrees well with COFIT, which is the important quantity. It should be mentioned that the central points are missing because of the large amplification of statistical scatter which occurs at 90° . These central points are used to provide the amplitude normalization, and are included in the COFIT input data. Points near the core boundaries and the aluminum plate are not entered into COFIT.

After analyzing the data in this way, it was found that both the top and bottom vertical reflector savings are constant for these water heights, as is the horizontal value. Bucklings for the three configurations are given in Table 7.5, the result for the 26 x 26 rod core is the most reliable because the water level was well below the aluminum plate.

E. Stability of Vertical Reflector Savings in a Core with Rod Absorbers

Vertical reflector savings values have been measured in various positions in a 2.72% Zr-4 clad core with uniform arrays of 25 Ag-In-Cd rods in the 0.600-inch lattice. The reflector savings is independent of position, and agrees with the result for clean cores.

Fuel rods were irradiated in the positions shown in Fig. 7.16 and counted. The data from rods in equivalent positions were averaged and analyzed for reflector savings. The results for each group are given in Table 7.6 for uniform arrays of 0.403" and 0.330" diameter Ag-In-Cd rods in the 31 x 31 core. The accuracy of each value of H_{ex} is about ± 0.5 cm, so no systematic variation was found with position. Also, the results are in reasonable agreement with those from clean cores.

F. Water Gap Measurements in a Zirconium Clad Core

Power production has been measured thru water gaps of various thicknesses in a core of 552, 2.72% Zr-4 clad fuel rods in the 0.600-inch lattice. Stainless steel and aluminum sheets were also inserted into the gap. Critical water heights were obtained for all cases.

The 0.600-inch pitch lattice plates were cut in the middle so that any size gap could be made in the center of a square core. Clamps were made to hold the plates at each desired spacing. Power production and critical water heights were measured for all conditions given in Table 7.7, and the power measurements were made at the positions indicated in Figure 7.19. Figures 7.17 and 7.18 show the power plots for effective slots of 0.060 and 0.120 inch respectively. Data points at ± 8.5 and ± 6.5 were normalized to the unperturbed curve, and an expanded power scale used to show the consistency of the plots.

TABLE 7.5

Buckling and Reflector Savings
of Square 2.72% Zr-4 Clad 3.1:1 W/U Cores

<u>Core Description</u>	<u>Critical Water Height (cm)</u>	<u>$B^2(M^{-2})$</u>
23 x 23	100.58	85.6 ± 1.4
24 x 24	75.48	86.0 ± 1.2
26 x 26	55.75	86.4 ± 1.1 (preferred value)

Horizontal Reflector Savings : $7.63 \pm .22$ cm

Vertical Reflector Savings = Bottom = $7.47 \pm .08$ cm
Top = $5.62 \pm .08$ cm

FIGURE 7.16

31 x 31, 2.7% Zr-4 Clad core with
25 Ag-In-Cd rods. Positions of
vertical power scans are indicated
by the numerals 1 thru 4.

TABLE 7.6

Vertical Reflector Savings
in 2.7% Zr-4 Clad Cores with
a Uniform Array of Ag-In-Cd Rods

Ag-In-Cd Rod Dia.	.403"	.330"
Critical Water Height	92.18 cm	69.70 cm
H _{ex} : Group 1	105.9	84.0
Group 2	106.0	84.0
Group 3	106.4	83.7
Group 4	105.4	83.7

TABLE 7.7

Critical Water Heights and Reactivities
for Water Gaps and Absorbers
in 552 Rod Zr-Clad Core with 0.600" Pitch

Effective Slot (ins)	Total Gap (ins)	Absorber	Critical Water Height (cm)	Reactivity (\$)
0.0	0.0	none	85.52	0.0
0.060	0.060	none	82.48	0.27
0.060	0.096	0.018" SS sheet	87.11	-0.13
0.060	0.134	0.037" SS sheet	90.94	-0.43
0.059	0.121	0.062" Al sheet	85.07	-0.04
0.121	0.121	none	79.79	0.54
0.120	0.156	0.018" SS sheet	84.31	0.11
0.120	0.194	0.037" SS sheet	88.51	-0.25
0.120	0.182	0.062" Al sheet	82.23	0.30

REACTIVITY POWER PRODUCTION THRU CORE
DISCONTINUITIES, EFFECTIVE SLOT 0.060 INCH

Figure 7.17

RELATIVE POWER

Figure 7.18

REACTIVITY POWER PRODUCTION THRU CORE
DISCONTINUITIES, EFFECTIVE SLOT 0.120 INCH

552, 2.72% ZIRCONIUM CLAD FUEL RODS IN 0.600-INCH LATTICE.
POWER PRODUCTION MEASURED IN INDICATED FUEL RODS.

Figure 7.14

Note that the 0.060-inch effective slot data, Figure 7.17, agree quite well with each other, as do the 0.120-inch data.

The reactivity values given in Table 7.7 were calculated using the expression given in A.

Perturbation Mock-up Critical Experiments
P. M. Wood, R. D. Leamer

A. Comparison of Experimentally Determined Power Peaking Near Water Slots with Analysis

A series of experiments to determine power peaking in fuel rods adjacent to water slots was carried out at the Westinghouse Reactor Evaluation Center.

The experimental configuration consisted of a 24×23 array of 2.72% enriched 0.484" O.D. Zircaloy-4 clad UO_2 fuel rods on a 0.600" pitch. The arrangement allowed one-half of the core (12×23) to be moved relative to the other half to provide a water slot through the center of the core. Experiments were performed with sheets of aluminum and of Type 304 stainless steel centered in the water slot. The cases considered are given in Table 7.8.

The analysis consisted of a PDQ-03 calculation using two-group constants obtained from the LEOPARD code. MND thermal constants were used. "Soft-spectrum" water constants were used in the reflector and water slot. For cases with aluminum or Type 304 stainless steel inserts homogenized "soft-spectrum" constants were used.

The water slot (or water plus insert) was represented by a single mesh spacing in the direction perpendicular to the slots. Four mesh spacings per fuel rod were used near the center and at the edge of the core with two mesh spacings per fuel rod in between.

The experimental and calculated peaking factors for the central fuel rods adjacent to the water slot are given in Table 7.8. The experimental value is the average of four rod scans. The off center row of rods was normalized to the central row using the PDQ results.

In general, PDQ overestimates the power peaking by 2-3%. The agreement is better with water and water plus Al than it is with water plus stainless steel.

The calculation underestimates the poisoning effect of Type 304 stainless steel by 0.1% in K_{eff} .

Table 7.8

Peaking Factors and Reactivity Effects
in Water Slot Experiments in the 2.72% Enriched 3.05 W/M Zircaloy Lattice

Total Gap (in)	Case	Critical Height (cm)	Experimental		Calculated	
	Metal Insert (in)		Peaking Factor	Reactivity* Change	Peaking Factor	Reactivity Change
None	None	85.52	1.0	0	1.0	0
0.060	None	82.48	1.023	+0.00211	1.032	+0.00236
0.121	0.062 Al	85.07	1.025	+0.00030	1.0465	+0.00036
0.096	0.018 SS	87.11	1.040	-0.00103	1.059	+0.00005
0.134	0.037 SS	90.94	1.027	-0.00331	1.054	-0.00253
0.121	None	79.79	1.062	+0.00415	1.088	+0.00453
0.182	0.062 Al	82.23	1.104	+0.00229	1.104	+0.00251
0.156	0.018 SS	84.31	1.075	+0.000815	1.109	+0.00169
0.194	0.037 SS	88.51	1.085	-0.00189	1.108	-0.00080

* $\beta_{\text{eff}} = 0.0077$

Figure 3.5
Saxton Reactor Plant Advanced Grid Design
Test Assembly (3x3)

THIS ASSY MUST PASS THRU A VERTICAL INSPECTION ENVELOPE GAUGE EQUAL IN LENGTH TO A FUEL ASSY, WITH THESE MAXIMUM CROSS SECTIONAL DIMS. WITHOUT RESTRICTIONS OR BINDING. A 1/8" THICK FEELER GAUGE MUST NOT PASS BETWEEN ENVELOPE GAUGE AND OUTSIDE SURFACE OF ENCLOSURE. (SEE NOTE 2)
(THIS NOTE TYP FOR ALL SECTION C-C)

[illegible]

PLANT ITEM NO.	NOTES:
503-3-4-21	<u>GROUP 1</u> - 3K3 TEST FUEL ASSY. WITH 5 THIN CLAD RODS WITH SAME ENRICHMENT AS CORE (4 REMOVABLE) # 4 ZIRC. RODS
503-3-4-22	<u>GROUP 2</u> - 3K3 TEST FUEL ASSY. WITH 5 THIN CLAD RODS WITH SAME ENRICHMENT AS CORE (4 REMOVABLE) # 4 ZIRC. RODS
503-3-4-1	<u>GROUP 3</u> - 2K2 TEST FUEL ASSY. WITH 4 3ST. RODS WITH SAME ENRICH. AS CORE (2 REMOVABLE)
503-3-4-23	<u>GROUP 4</u> - 3K3 ROC TEST FUEL ASSY. WITH 2 ROC ELEMENTS: 2 THIN CLAD RODS WITH THE SAME ENRICHMENT AS CORE, AND 5 SPIKED ZIRC. RODS
503-3-4-24	<u>GROUP 5</u> - 3K3 TEST FUEL ASSY. WITH 3 ZIRC CLAD RODS (2 DEFECTED), 5 VIBRATORY COMPACTED RODS (2 STAINLESS CLAD & 3 ZIRC CLAD) AND ONE REMOVABLE STAINLESS CLAD ROD
503-3-4-25	<u>GROUP 6</u> - 3K3 IMPROVED GRID DESIGN TEST ASSEMBLY WITH 5 3ST CLAD STANDARD RODS (4 REMOVABLE)
503-3-4-26	<u>GROUP 7</u> - 5K3 PLUTONIUM FUEL ASSEMBLY

NOTES

1. WELD PER (N) P.S. 292413-1. OMIT RADIOGRAPHIC INSPECTION.
2. INSPECTION ENVELOPE GAUGE & PROCEDURE TO BE APPROVED BY W.A.R.D.
3. RECORD PLANT ITEM NO. OF EACH ASSEMBLY IDENT. NO. & POSITION OF FUEL RODS, FUEL WEIGHT, AND METAL WEIGHT.

[illegible]