AI-CE-56

MASTER

HEAVY WATER ORGANIC COOLED REACTOR

QUARTERLY TECHNICAL PROGRESS REPORT OCTOBER THROUGH DECEMBER 1966

AEC RESEARCH AND DEVELOPMENT REPORT

ATOMICS INTERNATIONAL—COMBUSTION ENGINEERING

A Joint Venture for Heavy Water Organic Cooled Reactors

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Printed in the United States of America
Available from

Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia 22151

Price: Printed Copy \$3.00; Microfiche \$0.65

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

H.C. \$ 3.06; MIN 65

HEAVY WATER ORGANIC COOLED REACTOR

QUARTERLY TECHNICAL PROGRESS REPORT OCTOBER THROUGH DECEMBER 1966

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission
A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights, or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

Contract: AT(38-1)-430
FOR THE
U.S. ATOMIC ENERGY COMMISSION

ISSUED: AUG 15 1967

ATOMICS INTERNATIONAL-COMBUSTION ENGINEERING
A Joint Venture for Heavy Water Organic Cooled Reactors

DISTRIBUTION

This report has been distributed according to the category "Reactor Technology" as given in the Standard Distribution for Unclassified Scientific and Technical Reports, TID-4500.

CONTENTS

	Page
Reactor Physics (13200)	15
I. Project Objectives	15
II. Progress During Report Period and Evaluation of Work to Date	
A. CE-HAMMER Program	15
B. Collision Probability Methods.	17
C. Safe Monte Carlo Code	19
D. Analysis of SRL Experiments with $\text{PuO}_2\text{-UO}_2$ Fuel.	20
E. Uranium Carbide Experiments at Savannah River Laboratory	24
F. Resonance Escape	25
III. Next Report Period Activities	27
Demonstration Plant Engineering (14000)	29
Reference Plant Engineering (15000)	31
I. Project Objectives	31
II. Progress During Report Period and Evaluation of Work to Date . . .	31
A. General Studies	31
B. Reactor Design	37
C. Plant Engineering	43
III. Next Report Period Activities.	63
Dual Purpose Plant Engineering (16000)	65
I. Project Objectives	65
II. Progress During Report Period and Evaluation of Work to Date . . .	66
A. Phase B – Fuel Element Optimization	66
B. Phase C – Coupling Studies, Moderator Heat Usage Study	81
III. Next Report Period Activities.	85
Dual Stream Reactor Concept (16000A)	87
I. Project Objectives	87
II. Progress During Report Period and Evaluation of Work to Date . . .	87
A. Nuclear Analysis Methods	87
B. Nuclear Analysis Results	88
C. Fuel Utilization and Cost	93
D. Evaluation of Work to Date	93

CONTENTS

	Page
Pressure Tube (21000)	95
I. Project Objectives	95
II. Progress During Report Period and Evaluation of Work to Date .	95
A. Summary	95
B. Pressure Tube System Development	95
C. Single Channel Test Loop (TF-10)	105
D. Five Channel Test Loop (TF-20)	112
E. Component Test Loop (TF-50)	113
F. Pressure Tube Irradiation Tests	114
G. SAP Irradiations	117
H. Loop Coolant Chemistry and Leak Detection	117
III. Next Report Period Activities	119
Transition Joints (21200)	121
I. Project Objectives	121
II. Progress During Report Period and Evaluation of Work to Date .	122
A. SAP-Steel Hot-Pressed Transition Joints	122
B. SAP-Steel Rolled Joints	132
III. Next Report Period Activities	140
A. SAP-Steel Hot-Pressed Joints	140
B. SAP-Steel Rolled Joints	141
Pressure Tube Fabrication Development (21800)	143
I. Objectives	143
II. Progress During Report Period and Evaluation of Work to Date .	143
III. Next Report Period Activities	150
Reactor Control Systems (22000)	151
I. Project Objectives	151
II. Progress During Report Period and Evaluation of Work to Date .	151
A. General	151
B. Shutdown Control System	152
C. Xenon Oscillation and Power Level Control	156
D. Soluble Poison Control System	163
III. Next Report Period Activities	164

CONTENTS

	Page
Reactor Physics (13200)	15
I. Project Objectives	15
II. Progress During Report Period and Evaluation of Work to Date	
A. CE-HAMMER Program	15
B. Collision Probability Methods.	17
C. Safe Monte Carlo Code	19
D. Analysis of SRL Experiments with $\text{PuO}_2\text{-UO}_2$ Fuel.	20
E. Uranium Carbide Experiments at Savannah River Laboratory	24
F. Resonance Escape	25
III. Next Report Period Activities	27
Demonstration Plant Engineering (14000)	29
Reference Plant Engineering (15000)	31
I. Project Objectives	31
II. Progress During Report Period and Evaluation of Work to Date . . .	31
A. General Studies	31
B. Reactor Design	37
C. Plant Engineering	43
III. Next Report Period Activities.	63
Dual Purpose Plant Engineering (16000)	65
I. Project Objectives	65
II. Progress During Report Period and Evaluation of Work to Date . . .	66
A. Phase B – Fuel Element Optimization	66
B. Phase C – Coupling Studies, Moderator Heat Usage Study	81
III. Next Report Period Activities.	85
Dual Stream Reactor Concept (16000A)	87
I. Project Objectives	87
II. Progress During Report Period and Evaluation of Work to Date . . .	87
A. Nuclear Analysis Methods	87
B. Nuclear Analysis Results	88
C. Fuel Utilization and Cost	93
D. Evaluation of Work to Date	93

CONTENTS

	Page
Pressure Tube (21000)	95
I. Project Objectives	95
II. Progress During Report Period and Evaluation of Work to Date .	95
A. Summary	95
B. Pressure Tube System Development	95
C. Single Channel Test Loop (TF-10)	105
D. Five Channel Test Loop (TF-20)	112
E. Component Test Loop (TF-50)	113
F. Pressure Tube Irradiation Tests	114
G. SAP Irradiations	117
H. Loop Coolant Chemistry and Leak Detection	117
III. Next Report Period Activities	119
Transition Joints (21200)	121
I. Project Objectives	121
II. Progress During Report Period and Evaluation of Work to Date .	122
A. SAP-Steel Hot-Pressed Transition Joints	122
B. SAP-Steel Rolled Joints	132
III. Next Report Period Activities	140
A. SAP-Steel Hot-Pressed Joints	140
B. SAP-Steel Rolled Joints	141
Pressure Tube Fabrication Development (21800)	143
I. Objectives	143
II. Progress During Report Period and Evaluation of Work to Date .	143
III. Next Report Period Activities	150
Reactor Control Systems (22000)	151
I. Project Objectives	151
II. Progress During Report Period and Evaluation of Work to Date .	151
A. General	151
B. Shutdown Control System	152
C. Xenon Oscillation and Power Level Control	156
D. Soluble Poison Control System	163
III. Next Report Period Activities	164

CONTENTS

	Page
Reactor Physics (13200)	15
I. Project Objectives	15
II. Progress During Report Period and Evaluation of Work to Date	
A. CE-HAMMER Program	15
B. Collision Probability Methods.	17
C. Safe Monte Carlo Code	19
D. Analysis of SRL Experiments with $\text{PuO}_2\text{-UO}_2$ Fuel.	20
E. Uranium Carbide Experiments at Savannah River Laboratory	24
F. Resonance Escape	25
III. Next Report Period Activities	27
Demonstration Plant Engineering (14000)	29
Reference Plant Engineering (15000)	31
I. Project Objectives	31
II. Progress During Report Period and Evaluation of Work to Date . . .	31
A. General Studies	31
B. Reactor Design	37
C. Plant Engineering	43
III. Next Report Period Activities.	63
Dual Purpose Plant Engineering (16000)	65
I. Project Objectives	65
II. Progress During Report Period and Evaluation of Work to Date . . .	66
A. Phase B – Fuel Element Optimization	66
B. Phase C – Coupling Studies, Moderator Heat Usage Study	81
III. Next Report Period Activities.	85
Dual Stream Reactor Concept (16000A)	87
I. Project Objectives	87
II. Progress During Report Period and Evaluation of Work to Date . . .	87
A. Nuclear Analysis Methods	87
B. Nuclear Analysis Results	88
C. Fuel Utilization and Cost	93
D. Evaluation of Work to Date	93

CONTENTS

	Page
Pressure Tube (21000)	95
I. Project Objectives	95
II. Progress During Report Period and Evaluation of Work to Date .	95
A. Summary	95
B. Pressure Tube System Development	95
C. Single Channel Test Loop (TF-10)	105
D. Five Channel Test Loop (TF-20)	112
E. Component Test Loop (TF-50)	113
F. Pressure Tube Irradiation Tests	114
G. SAP Irradiations	117
H. Loop Coolant Chemistry and Leak Detection	117
III. Next Report Period Activities	119
Transition Joints (21200)	121
I. Project Objectives	121
II. Progress During Report Period and Evaluation of Work to Date .	122
A. SAP-Steel Hot-Pressed Transition Joints	122
B. SAP-Steel Rolled Joints	132
III. Next Report Period Activities	140
A. SAP-Steel Hot-Pressed Joints	140
B. SAP-Steel Rolled Joints	141
Pressure Tube Fabrication Development (21800)	143
I. Objectives	143
II. Progress During Report Period and Evaluation of Work to Date .	143
III. Next Report Period Activities	150
Reactor Control Systems (22000)	151
I. Project Objectives	151
II. Progress During Report Period and Evaluation of Work to Date .	151
A. General	151
B. Shutdown Control System	152
C. Xenon Oscillation and Power Level Control	156
D. Soluble Poison Control System	163
III. Next Report Period Activities	164

CONTENTS

	Page
Fuel (28000)	165
I. Project Objectives	165
II. Progress During Report Period and Evaluation of Work to Date . .	165
A. Fuel Materials	165
B. SAP Cladding Development	165
C. Fuel Assembly Fabrication Development	175
D. Irradiation Testing	181
E. Out-of-Pile Testing	203
F. Physics Fuel Production	213
III. Next Report Period Activities	215
A. SAP Development	215
B. Fuel Assembly Fabrication Development	215
C. Irradiation Testing	216
D. Out-of-Pile Testing	217
E. Physics Fuel Productions	217
Large Component Testing (31000)	219
Organic Coolant Evaluation and Analytical Techniques (35110)	221
I. Project Objectives	221
II. Progress During Report Period and Evaluation of Work to Date . .	221
A. Physical Properties	221
B. Decomposition and Makeup Rates – MIT In-Pile Loop Facility	221
C. Coolant Oxidation	222
D. Halogen Chemistry	223
III. Next Report Period Activities	225
Parameters Affecting Film Formation (35120)	227
I. Project Objectives	227
II. Progress During Report Period and Evaluation of Work to Date . .	227
A. Film Formation Loop	227
B. Thermal Coking Tests	228
C. U-3 Loop Film Formation Tests	228
D. Analysis of Film Formation Data	229

CONTENTS

	Page
III. Next Report Period Activities	230
A. Film Formation Loop	230
B. Thermal Coking Tests	230
C. U-3 Film Formation Tests	230
D. Evaluation of Fouling Data	230
Coolant Technology Loop (35700)	231
I. Project Objectives	231
II. Progress During Report Period and Evaluation of Work to Date . .	231
A. Loop Design	231
B. Coolant Technology Program	232
III. Next Report Period Activities	232
Coolant Purification and Recovery (67000)	233
On-Power Refueling (81000)	235
I. Project Objectives	235
II. Progress During Report Period and Evaluation of Work to Date . .	235
A. Refueling Machine Design	235
B. Fuel Transfer Equipment Design	239
C. Fuel Handling Machine Development and Testing	239
III. Next Report Period Activities	243
A. Refueling Machine Design	243
B. Refueling Machine Development	243

TABLES

13200-1. Comparison of CE-HAMMER and PIP-I Calculations	16
-2. Comparison of 4-Group Reaction Rates	18
-3. Isotopic Analyses and Densities	21
-4. Comparison Between SRL Experiments and Calculations for PuO_2 - UO_2 Fuel	22
-5. Case 1232, SRL Experiments, 19 Rods, 9.33-in. Pitch Fuel C (0.3 wt % Pu-239, 0.3 wt % U-235), HB-40 Coolant	23
-6. Tentative HWOCR Experimental Program at Savannah River Laboratory for FY 1967	26
15000-1. Natural UC Fuel Assembly Designs	40
-2. Summary of Physics Data for Natural UC Fueled HWOCR With 19-Rod Cluster Fuel Assembly	42

TABLES

	Page
15000-3. H ₂ Cost (\$/100 SCF) for Catalytic Hydrocracker	45
-4. Summary of Additional Direct Capital Costs for 100% Turbine Bypass Systems	46
-5. Main Heat Transfer System Design Data 1000 Mwe Natural UC Fueled HWOCR	60
-6. Turbine Plant Design Data 1000 Mwe Natural UC Fueled HWOCR	62
16000-1. Parameters Employed in Physics Analyses of Three Principal Fuel Systems	67
-2. Primary Heat Transfer Loop Study Cases	70
16000A-1. Beginning of Life Reactivity Results for an Enrichment of 1.5 wt % U ²³⁵	89
-2. Fuel Inventory Data for 20,000 Mwd/MTU Exposure	91
-3. Reactivity Coefficients	92
-4. Fuel Utilization and Cost	94
21000-1. TF-20 Pressure Tube Development Tests	96
-2. TS 10-4 Operation	112
-3. SAP Pressure Tube Dimensions	115
-4. TF-50 Coolant Chemistry Sample Analysis	118
-5. B. F. Goodrich and CANDU Coke Analysis	119
21200-1. Summary of Tensile Data for Thermal-Cycled SAP-Steel Transition Joints	122
-2. Tensile Test Data for Flash-Welded SAP-895 4.000-in. ID x 0.100-in. Wall Tubing	124
-3. Tensile Test Data for 0.350-in.-Diameter Rod-Type Transition Joint Control Specimens for WR1-953 and -954 . .	124
-4. Correlation of Nondestructive with Destructive Tungsten Thickness Values	129
-5. Tests of Rolled Joints	134
21800-1. SAP Pressure Tube Inspection Summary (Lot No. 1)	145
-2. Tensile Data on SAP 895 Pressure Tubes (Lot No. 1) (All Tests at Room Temperature on Flat Strips Cut Longitudinal to Tube Axis)	146
-3. Hoop Data on SAP 895 Pressure Tubes Using Ring Specimens (Lot No. 1).	147

TABLES

	Page
21800-4. SAP Tubing Dimensional Measurements (Lot No. 2)	148
-5. Ultrasonic Inspection of SAP Pressure Tubes (Lot No. 2)	148
-6. SAP Pressure Tubing Dimensional Data for WR1 Irradiation Experiment (Lot No. 3)	149
-7. SAP Pressure Tube Extrusion Parameters (Lot No. 3)	149
-8. Stress Rupture Tests of SAP 895 Pressure Tubing	150
28000-1. Montecatini Billet Receivals and Schedule	166
-2. Alcoa XAP 005 Billet Receivals and Schedule	166
-3. Aluminum Oxide Content of Alloys and Billets	168
-4. Impurity Compositions of SAP 895 and XAP 005 Billets (ppm)	169
-5. Inclusion Count Data on Billets	169
-6. Ultrasonic Test of 2-in. Diameter Montecatini SAP 895 Alloy Billet	173
-7. Ultrasonic Test of 7-in. Diameter Montecatini SAP 895 Alloy Billets	173
-8. Mechanical Properties Data of Alloys and Billets	174
-9. Tube Burst Stress Rupture Tests of Cladding	174
-10. MFW Parameters for Piqua Fuel End Closures	176
-11. Tensile Strength of Piqua MFW End Closures at 850°F	176
-12. Compression Tests on Large Piqua Cladding (XAP 005) with and Without MFW End Closures	177
-13. B Rod Density and Dimensional Data	186
-14. H Rod Density and Dimensional Data	187
-15. J Rod Density and Dimensional Data	188
-16. Range of Experiment Performance September 15- December 15, 1966.	192
-17. EXP-WR1-951 Operating Status	199
81000-1. Snout Seal Test Results	238
-2. Static Face Seal Data	238
-3. Organic Soak Performance (In Santowax OM at 400°F)	240
-4. Organic Bake Performance (In Air at 400°F)	240

FIGURES

	Page
15000-1. Pump Turbine System Inertia vs Required Isolation Valve Closure Time (Gate Valves)	32
-2. Pump Drive Shaft Shear Reactor Coolant Flowrate vs Time	34
-3. Steam Generator Shell Rupture Reactor Coolant Inlet Temperature vs Time	34
-4. Natural Uranium Studies (19-Rod Design)	38
-5. Simplified Block Diagram of HWOCR Primary System	47
-6. Reactor Coolant Flowrate Transients – Loss of One Coolant Loop by Pump Tripout	48
-7. Reactor Coolant Flowrate Transients – Loss of One Coolant Loop by Pump Trip and Valve Closure	49
-8. Reactor Coolant Flow Transients – Loss of One Coolant Loop by Pump Trip or by Valve Closure	49
-9. Pump-Turbine Inertia vs Reactivity Withdrawal Rate for Reactor Power Level Control in Event of Loss of Flow in One Coolant Loop	51
-10. Reactor Coolant Temperature Program	53
-11. Net Plant Efficiency	54
-12. Net Plant Output and Main Turbine Flow.	54
-13. Reactor Fission Power Coolant Thermal Power.	55
-14. Evaporator Thermal Power, Preheat Section	55
-15. Evaporator Thermal Power, Boiling Section.	56
-16. Evaporator Thermal Power, Preheat and Boiling Sections. .	56
-17. Evaporator Outlet Steam Pressure and Temperature	56
-18. Feedwater Temperature	56
-19. Superheater Thermal Power.	57
-20. Reheater Thermal Power.	57
-21. Reheater Temperature Requirements.	57
-22. 1000 Mwe HWOCR Plant Improvement in Net Plant Efficiency Due to Utilization of D ₂ O for Feedwater Heating as a Function of Condenser Pressure	58
-23. Reactor Plant Heat Balance, 1000-Mwe HWOCR Plant, Natural Uranium Core (UC Fuel)	59
-24. Steam and Feedwater System Heat Balance, 1000 Mwe HWOCR Plant, Natural UC Core.	61

FIGURES

	Page
16000-1. Primary Loop Cost vs Reactor ΔT	72
-2. Relative Water Costs — Indirect and Direct Cooling System	82
-3. Typical Schematic Diagram of Heat Pump Principle Cycle (Case 6)	84
21000-1. Laboratory Wear Test Setup.	98
-2. Wear Test — Various Specimens on SAP 895 Tubing	99
-3. Shield Plug Pressure Drop	100
-4. TF-20 Shield Plug.	101
-5. Pressure Tube End Seal With Thermal Baffle Assembly	103
-6. Temperature Distribution of Thermal Baffle Assembly	104
-7. Groove Inspection of Inside Diameter of TS 10-4	106
-8. TS 10-4 Wear Marks on Pressure Tube ID as Viewed Through Borescope.	107
-9. TF 10-4 Fuel Bundle Vibration Amplitude vs Distance from Bottom of Bundle (Bundles Aligned)	109
-10. Vertical Motion of Magnet No. 17 in TF-10 Fuel String	109
-11. TS 10-4 SAP-to-SAP Flash Weld Failure as Viewed Through Borescope	110
-12. TS 10-4 SAP-to-SAP Flash Weld Failure	111
-13. TF-20 Control Room	113
21200-1. TS 20-4 Pressure Tube Test Section	123
-2. Present and Alternate Concepts of the Pressure Tube Subassembly	126
-3. Chemically Milled XAP 001 Pressure Tube Test Specimen.	128
-4. Test Instrumentation, Fixturing, and Samples used to Demonstrate the Feasibility of Using Hall Effect to Measure Electroplated Nickel Layer Thickness on Pres- sure Tube Transition Joint Section	130
-5. Probe Assembly Containing Hall Device in Contact With Nickel-Plated "Knife Edge" on Stainless-Steel Section of Pressure Tube Transition Joint	130
-6. Hall Device Scans Around Periphery of 347 SS Joint Sections Illustrating Effectiveness of Method to Deter- mine Electroplated Nickel Layer Thickness	131

FIGURES

	Page
21200-7. Summary of Rolled Joint Models	133
-8. Double-Ended Model 15 and Model 17 Rolled Joints Showing Type of Failure After Tensile Testing at 750°F. (Failure occurred at a stress level of about 15,000 psi)	135
-9. Cross Section of a Model 15 Rolled Joint	135
-10. Interface Between SAP Tube and Inner Stainless Steel Sleeve. (Oxide layer acted as effective diffusion barrier).	136
-11. Intermetallic Formation at Interface Between SAP Tube and the Inner Stainless Steel Sleeve Near Closed End of Joints	136
-12. Cross Section of Model 17 Rolled Joint.	137
-13. Characteristic Growth Patterns of the Intermetallics at the Interface Between the Outer SS Sleeve and the SAP Tube	138
-14. Closeup of Intermetallic Shown in Figure 21200-11.	139
22000-1. Load vs Current for Double Coil Magnet, Both Coils Energized.	153
-2. Load vs Current for Double Coil Holding Magnet, Rear Coil Energized.	154
-3. Load vs Current for Double Coil Holding Magnet, Front Coil Energized	154
-4. Horizontal Power Control Rod Schematic (Showing One Section of the Tri-Foil Arrangement).	158
-5. Vertical Support and Instrumentation Column, Power and Xenon Control System	159
28000-1. Metallic Inclusion in Alcoa 2-in. Diameter XAP 005 Billet (Typical in 0.0010 to 0.0030 in. range. View is of transverse section from front of billet No. 41-6) (100X)	171
-2. Metallic Inclusion in Alcoa 2-in. Diameter XAP 005 Billet (Typical in 0.0030 in. range. View is of trans- verse section from rear of billet No. 59-B) (100X)	171
-3. Large (0.11 by 0.015 in.) Metallic Inclusion in Sample from Billet No. 45-1 (Alcoa 7-in. diameter XAP 005) (100X)	171
-4. Rear View of Discontinuities on Cross Section of Billet No. 45-1 (Alcoa 7-in. diameter XAP 005. Actual Size).	171

FIGURES

	Page
28000-5. Discontinuity in Rear End of 2-in. Diameter SAP 895 Montecatini Billet No. 557. (Light areas are uneven etch regions. Actual size.)	171
-6. Original and New Collet Designs	178
-7. Components for Cold-Headed End Plug, Joint Trials	180
-8. Cross Section of Cold-Headed End Plug (5X)	180
-9. U-305 Bundle LTA Rod B Typical Fuel and Cladding Appearance	183
-10. U-305 Bundle LTA Rod H Typical Fuel and Cladding Appearance	184
-11. U-305 Bundle LTA Rod J Typical Fuel and Cladding Appearance	185
-12. U-305 Bundle LTA Rod B Typical Metallography (250X etched) (Reaction is into fuel approximately 1 mil - not into cladding)	189
-13. U-305-A1 Flow-Delta P Tests	192
-14. Flow-Delta P Test on Dummy Bundle	192
-15. Overall Hydraulic History of U-305-A1	194
-16. (Part 1). Thermal History of U-305-A1	195
-16. (Part 2). Thermal History of U-305-A1	196
-17. EXP-WR1-951 Operational History	198
-18. EXP-WR1-951 Hydraulic Tests	199
-19. Knoop Hardness vs Location Across UC-SAP Interface, Capsule DH-70 (150X)	202
-20. Gradient Microstructure Next to the "Major" Reaction of the 5.3 wt % C in UC-SAP Interface of Capsule DH-70 (150X)	203
-21. Knoop Hardness Location Across UC-SAP Interface, Capsule DH-87 (150X)	204
-22. Knoop Hardness Location Across UC-SAP Interface, Capsule DH-74 (150X)	206
-23. Material Adhering to Surface of XAP 001 Cladding from NRU-305 Irradiation Test (~0.001-in. thick; as polished) (200X)	207
-24. Small Spherical Fuel Particle (~0.002-in. diameter) Embedded in the Cladding from NRU-305 Irradiation Test (500X)	207

FIGURES

	Page
28000-25. Material Appended to SAP Cladding Surface with Included Particulate Matter (From NRU-305 Irradiation Test) (500X).	208
-26. Intrusion of SAP Spike into Hyperstoichiometric UC of Capsule DH-83. (Interface between SAP spike and UC shows reaction zone. Capsule heat-treated 2000-hr at 850°F) (1000X).	209
-27. Hyperstoichiometric UC/SAP Compatibility Test Data (Capsules DH-1 through DH-88; status on 12-16-66).	210
-28. Instrumented Subchannel Locations	212
-29. Axial Dye Distribution for a 2-String Fuel Assembly (Subchannel K).	212
-30. Axial Dye Distribution for a 2-String Fuel Assembly (Subchannel A).	213
81000-1. CANDU Mechanism Sliding Surface After Baking in Santowax-OM.	242

1

2

3

4

5

HWO CR QUARTERLY REPORT	
TITLE: Reactor Physics	WBS NO: 13200
REPORTING PERIOD: October through December 1966	

I. PROJECT OBJECTIVES

The objectives of the reactor physics task are to provide nuclear design methods incorporating digital computer codes, nuclear cross-section data and analysis techniques which will predict the nuclear characteristics of HWO CR plants with sufficient accuracy. The uncertainties in such nuclear characteristics as fuel enrichment, lifetime power distribution, control requirements, plutonium production, and reactivity coefficients must be minimized to achieve low-energy costs.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. CE-HAMMER PROGRAM

The original HAMMER^{*} code, which was adequate for analysis of simple lattice criticals, has been under major revision to make it into a more useful design code for HWO CR. Long running time for regionwise resonance integral calculations, reported in the previous progress report,[†] has been halved with the incorporation of a 6-point interpolation on collision probabilities as a function of energy. An option also has been provided whereby the cell averaged resonance integral may be used in all fuel regions to reduce running time further in obtaining regionwise cross sections. Edits of power distributions, peaking factors, regional flux integrals and four-group, regionwise macroscopic cross sections for design use have been added. To provide better data for experimental analysis, material activation edits have been improved and epithermal effects have been incorporated.

^{*}J. Suich and H. Honeck, "The HAMMER Code," DP 975 (1965).

[†]"Heavy Water Organic Cooled Reactor Quarterly Progress Report, August through September 1966," AI-CE-55.

A 4-group regionwise burnup module which has been written for CE-HAMMER is now being tested. This option (modeled after CINDER)* allows generalized linear transmutations represented by chains of 15 nuclides or less. Up to 200 nuclides from an external few-group cross-section library can be used in addition to the CE-HAMMER isotopes. A 4-group adaptation of the PIP-I† integral transport code with a buckling search is used at intermediate time steps to maintain criticality and to recalculate power distributions and few-group fluxes. An option is now being added for performing a full 84-group HAMMER calculation at major time steps to recalculate the neutron spectrum and effective cross sections.

As part of the testing, a comparison between the 84-group integral transport calculation in CE-HAMMER and the four-group integral transport calculation used in the burnup module has been made for a typical HWO CR lattice of 37 rods with six square inches of fuel and a deuterium/uranium (D/U) atom ratio of 27.5. A summary of eigenvalues and power distributions is given in Table 13200-1.

TABLE 13200-1
COMPARISON OF CE-HAMMER AND
PIP-I CALCULATIONS

	CE-HAMMER	PIP-I
Relative Power Distribution		
Center Rod	0.019	0.018
Inner Ring	0.127	0.122
Middle Ring	0.314	0.313
Outer Ring	0.539	0.546
k_{eff}	1.194	1.211
Third Group Escape Probability	0.823	0.842

*T. R. England, "CINDER - A One Point Depletion and Fission Product Program," WAPD-TM-334 (August 1962)

†A. Clayton, "The Programme PIP-I for the Solution of the Multigroup Equations of the Method of Collision Probabilities," AEEW-R-326 (March 1964)

Power distributions are in very good agreement. The eigenvalue determined by the four-group calculation is, however, about 1.4% higher than that calculated using HAMMER. This eigenvalue discrepancy occurs largely because of two factors: (1) differences in flux distributions predicted by multigroup and few broad group representations, and (2) an additional flux dip in the explicitly represented resonance fuel region in PIP-I, as is shown by the difference in resonance escape probability in Table 13200-1.

A further comparison between the HAMMER and PIP-I calculations was carried out using a homogenized cell representation; in this case, the four-group and multigroup eigenvalues were the same. One objective of this comparison was to explore the consequence of using two different methods of evaluating the fine group resonance absorption cross sections. It was noted that the alternate scheme used in GAM^{*} of defining fine group resonance absorption cross sections as the resonance integral divided by the fine group lethargy width instead of maintaining the ratio of slowing down to absorption could increase eigenvalues by nearly a percent.

B. COLLISION PROBABILITY METHODS

As an aid in testing the validity of using an annular representation in physics design codes instead of an explicit rod cluster geometry, two collision probability codes developed in foreign laboratories have been made operational to compare explicit and annular representations. The Swedish collision probability code CLUCOP[†] uses a network of parallel lines for various angular orientations and assumes isotropic scattering. Comparisons with the Monte Carlo code TRAM[§] indicate good agreement. For a 37-rod cluster with fuel, coolant, and moderator absorption cross sections of 0.3, 0.486, and 0.221 cm⁻¹, respectively, the first collision probability in the fuel for a uniform source in the fuel was 0.355 using TRAM and 0.361 using CLUCOP. Further work on CLUCOP is required to debug an option allowing subdivision of annular regions by radial lines and to eliminate a relatively unimportant error in treating some cladding annuli.

*G. D. Joanou and J. S. Dudek, "GAM-I; A Consistent P₁ Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants," GA-1850, (June 28, 1961)

†I. Carlvik, "A Method for Calculating Collision Probabilities in General Cylindrical Geometries and Application to Flux Distributions and Dancoff Factors," A/Conf. 28/P/681 (May 1964).

§R. A. Pfeiffer and W. W. Stone, "TRAM for the Philco-2000, KAPL-M-RPC-1 (March 24, 1961).

The British code PIP-I which utilizes collision probabilities and neutron energy transfer matrices is used for few-or-multigroup, multiregion reactivity calculations. A partial test between geometrical representations was made for a 37-rod cluster with six square inches of fuel, on a ten-inch square lattice pitch. Four-group, regionwise microscopic cross sections from an annular HAMMER calculation were used in four-group PIP-I calculations for the explicit rod cluster geometry and the annular geometry employed in the HAMMER calculation. Table 13200-2 shows neutron balances in each energy group. The power distribution for the explicit rod cluster geometry is significantly flatter than for the annular representations; the relative power densities are 0.829 and 0.672, respectively, for the center rod, and 1.144 and 1.245, respectively, for the smaller outer rods.

This comparison does not take into account possible differences in regionwise 4-group cross sections which may result from an 84-group solution to the space-energy problem as embodied in HAMMER using the explicit rod cluster geometry. The explicit cluster eigenvalue was 0.27% lower than for the annular representations.

Further assessments using larger numbers of fine groups will be made.

TABLE 13200-2
COMPARISON OF 4-GROUP REACTION RATES*

Group	Geometry	Fission Neutrons Produced	Absorptions	Removal to Next Lower Group
1	Cluster	0.0543	0.0236	0.5834
2	Cluster	0.0036	0.0197	0.7645
3	Cluster	0.0439	0.1225	0.6414
4	Cluster	0.8982	0.6414	-
1	Annular	0.0528	0.0229	0.6173
2	Annular	0.0036	0.0195	0.7799
3	Annular	0.0413	0.1160	0.6471
4	Annular	0.9022	0.6471	-

*Normalized to one fission neutron produced.

C. SAFE MONTE CARLO CODE

The modifications and expanded output incorporated into SAFE have been debugged and checked against previous calculations of k_{∞} , ϵ , η , p , and f to ensure against introduction of errors into previously debugged portions of the code. The results were also compared with previously reported RBU results to further corroborate the validity of the procedures used in SAFE. In all comparisons performed, the SAFE results appeared reasonable and in sufficient agreement to assume their validity.

Several short runs were made in which the extent to which Russian Roulette and statistical weighting techniques were employed was varied. The statistical quality of these results was then evaluated in an attempt to optimize the use of these techniques. While assurance of optimization is difficult, the quality of the best set of these results is quite encouraging. A value of k_{∞} of 1.306 ± 0.0146 was obtained in 11 minutes and 47 seconds of computation time on the 7094. Quantities such as p , η , ϵ , and f were much better behaved, while region dependent fluxes appeared converged to better than 5% probable error.

A statistical estimation procedure was developed to more adequately treat both absorption and scattering processes in thin regions such as the pressure tube, calandria tube, and fuel cladding. Its success was verified by the fact that the statistical quality of thin region results was comparable to that in the large regions such as the moderator.

Four-thousand-history SAFE runs were made of the explicit geometric representation of the HWOCR cell and of the annular representation. These results were then compared with the annular representation of the cell computed by HAMMER. Agreement to 1% or better on k_{∞} , p , ϵ , η , f between the HAMMER calculation and the equivalent annular representation calculated by SAFE gives significance to those differences that are observed both between SAFE and HAMMER and between the annular and explicit representations calculated using Monte Carlo.

The work previously described was performed on a 7094 machine. The 7094 version of HWOCR-SAFE has been converted to the IBM 360. While this 360 version of SAFE is now mechanically operational on the 360, access to the

computer has been too limited to permit a detailed comparison with the same calculation performed with the 7094 version. Such a comparison is necessary because the 360 carries one and one-half less significant digits in its computations than does the 7094. This causes excessive round-off error in some tabulations and computations performed in SAFE. In addition, it required the development and testing of a new random number generator. Therefore, it is not possible to reproduce an exact replica of the 7094 calculation but instead a several-thousand-history run will be made and converged results will be compared.

D. ANALYSIS OF SRL EXPERIMENTS WITH PuO_2 - UO_2 FUEL

The Savannah River Laboratory has recently conducted a series of experiments for the USAEC-AECL Cooperative Program using oxide fuel of both natural enrichment and of depleted uranium with added plutonium to simulate the characteristics of partially burned-up fuel. The fuel types and compositions used in these experiments are given in Table 13200-3. Preliminary experimental results were reported by J. L. Crandall (SRL).*

Selected lattices with HB-40 or air coolant were analyzed with the CE-HAMMER code for preliminary comparisons with experiment. Table 13200-4 provides a summary of the results. In general, the results are consistent with previously reported analyses of ZED-2 criticals, viz. with organic coolant present, the analytical model underestimates the eigenvalue by 1 to 2%.

One aspect of the analytical model which was examined with one lattice of PuO_2 - UO_2 fuel is the annular representation of the fuel cluster in the one-dimensional HAMMER code. This lattice was analyzed with CE-HAMMER using both the CE and SRL rod cluster representations. The SRL ring model uses a tighter nesting of the fuel annuli and, for this particular fuel channel geometry, has a separate ring of pure coolant at the inside of the pressure tube. The results are summarized in Table 13200-5.

The prediction of k_{eff} is not as close to unity as Table 13200-5 would indicate. For example, the D_2O purity was taken as 99.75%, but the actual purity was about 99.55%, which reduces k_{eff} by 0.8%. Also, impurities in the fuel were ignored for this comparison, although actual chemical analyses showed

*Letter from J. L. Crandall (SRL) to A. G. Ward (AECL), October 31, 1966.

TABLE 13200-3
ISOTOPIC ANALYSES AND DENSITIES

Type of Material		Pu-wt % ⁽¹⁾				U wt % ⁽¹⁾			
		239	240	241	242	234	235	236	238
Type A	0.00295 Pu/U ⁽²⁾ $\rho = 10.421 \text{ g/cc}$	77.91	19.35	2.43	0.31	0.0012	0.302	0.0029	99.69
Type B	0.00259 Pu/U ⁽²⁾ $\rho = 10.356 \text{ g/cc}$	93.00	6.17	0.79	0.04	0.0013	0.298	0.0032	99.69
Type C	0.00340 Pu/U ⁽²⁾ $\rho = 10.380 \text{ g/cc}$	93.19	6.09	0.69	0.03	0.0012	0.301	0.0030	99.69
Type D	Depleted U $\rho = 10.403 \text{ g/cc}$					0.0026 ⁽³⁾	0.4979	0.0033	99.4962
E Rods	Natural U $\rho = 10.390 \text{ g/cc}$					0.0057	0.71	-	99.28

Impurity Analyses - ppm⁽⁴⁾

	Al	B	Ca	Cd	Cr	Cu	Fe	Mg	Mn	Mo	Na	Ni	Pb	Si	Sn	V
Type A	40	0.4	<10	<5	175	5	700	5	15	<10	50	20	<5	50	<10	<25
Type B	10	0.7	<10	<5	12	4	75	4	5	<10	100	15	<5	75	<10	<25
Type C	15	8	<10	<5	15	3	130	2	5	<10	75	20	<5	60	<10	<25

- (1) SRL Isotopic Analyses. Average of 2 samples
(2) SRL Pu/U Ratio by Weight
(3) ORNL Analysis
(4) SRL Analysis

TABLE 13200-4
COMPARISON BETWEEN SRL EXPERIMENTS AND
CALCULATIONS FOR $\text{PuO}_2\text{-UO}_2$ FUEL

Fuel	B	E	E	C	C	E	E
Rods per Channel	19	19	19	31	31	31	31
Lattice Pitch (in.)	9.33	9.33	9.33	9.33	12.12	9.33	12.12
Coolant	HB-40	HB-40	Air	HB-40	HB-40	HB-40	HB-40
$B^2(\text{m}^{-2})$							
Expt.	2.35	-	5.52	4.62	3.00	1.90	1.35
Calc.	1.21	-	5.28	3.77	2.77	0.78	0.96
$\rho^{28(a)}$							
Expt.	0.332	0.394	0.367	0.566	-	0.451	-
Calc.	0.3439	0.345	0.337	0.619	0.462	0.499	0.379
$\delta^{28(b)}$							
Expt.	0.048	0.045	0.050	0.065	-	-	-
Calc.	0.0459	0.0454	0.048	0.068	0.065	0.055	0.053
Lu/Mn Activation at 0 Radius							
Expt.	1.282	1.323	1.289	1.356	-	1.395	-
Calc.	1.296	1.339	1.314	1.399	1.360	1.428	1.390
$n, 1/v$ Activation at 4.6-in. Radius ^(c)							
Expt.	3.22	3.17	2.36	4.67	-	4.08	-
Calc.	3.14	3.10	2.37	4.53	5.50	3.88	4.70
Calc.							
k_∞	1.033	1.061	1.172	1.074	1.088	1.016	1.032
k_{eff}	0.9707	-	0.992	0.984	0.993	0.978	0.988
ρ	0.8735	0.878	0.871	0.818	0.857	0.828	0.864
η	1.2829	1.316	1.316	1.382	1.376	1.312	1.314
f	0.8607	0.857	0.951	0.870	0.836	0.852	0.838
Abs Boron ^(d)	0.0068	0.0	0.0	0.0020	0.0020	0.0	0.0
Boron (ppm) ^(e)	(3.18)	(0.0)	(0.0)	(1.0)	(1.0)	(0.0)	(0.0)
Abs. H_2O	0.0203	0.0208	0.0185	0.0068	0.0142	0.0074	0.0151
H_2O in D_2O (%)	(0.45)	(0.45)	(0.45)	(0.25)	(0.25)	(0.25)	(0.25)

(a) $\rho^{28} = \frac{\text{epithermal captures in U-238}}{\text{thermal captures in U-238}}$

(b) $\delta^{28} = \frac{\text{total fission in U-238}}{\text{total fission in U-235}}$

(c) $n(1/v)$ = thermal neutron density at $R = 4.6$ in., relative to unity at $R = 0$, as measured by activity of a $1/v$ absorber, or as calculated in HAMMER-THERMOS.

(d) Absorption in boron or H_2O is relative to unity for total absorption in lattice cell.

(e) Boron equivalent impurities (ppm):
Zero values used where no chemical analysis is available;
Nominal value of 1.0 ppm used for type C fuel due to uncertainty boron content.

TABLE 13200-5

CASE 1232, SRL EXPERIMENTS, 19 RODS, 9.33-in. PITCH
FUEL C (0.3 wt % Pu-239, 0.3 wt % U-235),
HB-40 COOLANT

	Expt. Result	CE HAMMER SRL Ring Model	CE HAMMER CE Ring Model
k_{eff}	1.00	0.9975	1.0041
k_{∞}		1.1221	1.1268
ρ		0.8704	0.8688
η		1.3793	1.3798
f		0.8732	0.8793
B_{m2}	~ 4.50	4.40	4.665
Lu/Mn	1.278	1.335	1.313
ρ^{28}	0.432	0.4066	0.4141
δ^{28}	0.056	0.0563	0.0551

impurities equivalent to about 3-ppm boron in several fuel samples (over 10 ppm in one fuel sample), which would further reduce k_{eff} by 0.2% per ppm boron. Taken together, the use of the most accurate analyses of D_2O and fuel impurities would give a value of k_{eff} about 1.4% less than unity. This discrepancy is about the same magnitude as found in analyses of organic-cooled cases in ZED-2 experiments. A reduction of 1% in k_{eff} could be obtained by a reduction of 0.01 wt % Pu-239, or 0.012 wt % U-235, showing the effect of small uncertainties in fuel composition.

The measured activation ratios for Lu-176 and Mn-55 are used as indicators of the thermal flux spectrum. In these experiments, the sub-cadmium activation was used for direct comparison with the HAMMER-THERMOS results. Typical experimental values of these ratios are about 1.30 in the central fuel rod, normalized to unity in a pure Maxwellian spectrum at moderator temperature. The ratios obtained from HAMMER with the CE ring model are generally about 3% high, and the tighter nesting of the fuel annuli increases the ratio by an additional 2%.

The shape of the thermal neutron density distribution in the lattice cell was measured using 1/v absorber foils and is indicated at the bottom of Table 13200-4, where the ratio of neutron density at the cell boundary to that at the center is given. The HAMMER neutron densities using the CE ring model were in quite good agreement with the experimental values.

Foil activation measurements with U-238 foils also gave values shown in Tables 13200-4 and -5 for the ratios ρ^{28} and δ^{28} averaged over all the fuel in a lattice cell. The calculated ratios were obtained from the HAMMER four-group edit of reaction rates. They are in reasonably good agreement with experiment. The approximate discrepancies range from 0 to 10% in ρ^{28} , mostly on the high side, and from 0 to 5% in δ^{28} . The values for ρ^{28} are decreased by approximately 2% for the SRL ring model, and may be increased by a similar amount if impurities are included.

E. URANIUM CARBIDE EXPERIMENTS AT SAVANNAH RIVER LABORATORY

Discussions were held with Savannah River Laboratory* regarding problems and planning for the critical experiments with uranium carbide fuel. Because of advantages of cost, procurement, and scheduling, the pressure tubes will be 6061-T6 aluminum in the standard commercial size of 4.00-in. ID, 0.125-in. wall. A stress analysis report† indicates that this thickness of aluminum is adequate for operation at 500°F, but not for 600°F; thus, the hot experiments will be limited to the lower temperature.

The fuel fabrication is proceeding at the new facilities of Atomic International. The uncertainty about the enrichment has been settled by several analyses which give the enrichment from all batches as $1.00 \pm 0.01\%$ as specified. The fuel rod specifications have been reviewed and issued.§ With the present schedule for delivery of fuel to SRL, the experimental program will begin early in April and most of it can be completed by the end of June.

*F. G. Moote, "Trip Report, Visit to Savannah River Laboratory, October 10-October 11, 1966," dated October 24, 1966, and F. G. Moote, "Trip Report, Visit to Savannah River Laboratory, November 30-December 1, 1966," dated December 5, 1966.

†H. Oslick, "Pressure Tube Analysis for SRL Experiments," TI-010-29-601 (AI-CE), November 15, 1966.

§"Specification - Fuel Rods and UC Slugs, HWOCR-SRL Experiments," AI-CE No. ST010N28001 - November 14, 1966.

A tentative program for FY 1967 is shown in Table 13200-6. A majority of the room temperature experiments are shown first on the schedule in the event that difficulties or delays with the elevated temperature equipment are encountered. Experiments listed under items A, B, C, and D are planned for the SE (subcritical experimental) facility. Item E, a verification of buckling by substitution technique for the reference lattice pitch (10 in.) and coolant (Dowtherm) at room temperature, is shown for the PDP facility. This particular experiment may not be carried out in the FY 1967 program since the present schedule for the PDP facility does not indicate the facility will be available in time to reload the core (approximately one month) and complete the experiment (approximately one week) prior to June 30, 1967.

F. RESONANCE ESCAPE

The discrepancy between REPLICA* and HAMMER resonance calculations particularly for the 6.7 ev U-238 resonance discussed in the previous progress report is being investigated further.

Cross sections equivalent to those in the resonance region in the present CE-HAMMER epithermal library have been placed in REPLICA. This code will be used to compare the effect of HAMMER resonance treatment on selected 19-rod cluster critical experiments.

An elastic transfer code ELTRA which prepares punched cards for the input of energy transfer matrices to PIP-I has been programmed and is being debugged. This will be used in 26-group PIP-I calculations using both annular and explicit geometries with CLUCOP collision probabilities to test the narrow resonance approximations in the coolant for the lowest energy U-238 resonance. It is possible that better resonance approximations for HAMMER can be devised using these methods.

*R. A. Dannels and S. M. Hendley, "REPLICA, A Monte-Carlo Code to Calculate Resonance Captures in a Lattice of 19-rod Hexagonal Clusters," CVNA-130 (May 1962).

TABLE 13200-6

TENTATIVE HWO CR EXPERIMENTAL PROGRAM* AT
SAVANNAH RIVER LABORATORY FOR FY 1967

A. Cold Experiments

1. Buckling as a function of lattice pitch for pitches of 9, 10, and 11 in.
2. Control element measurements
 - a. Reactivity worth and flux (power) distributions for one or two types of control rods and a light-water column oriented parallel to fuel channels; data measured for each control element separately.
 - b. Flux (power) distributions for above control rods oriented transverse to fuel channels.
3. Fuel bundle interface effects – measure flux (power) distributions as a function of fuel-to-fuel gap distance to define local axial peaking at ends of fuel bundles.

B. Isothermal Lattice Temperature Coefficient Measurement – change in buckling as whole lattice is heated from 20 to 80°C.

C. Cold (20°C) versus Hot (260°C) Measurements[†]

1. Fuel parameters: B^2 , ρ^{28} , δ^{28} , δ^{25} , and CR.
2. Spectral indices: Lu-176/Cu, Pu-239 (fission)/U-235 (fission).
3. Activation flux traverses (Mn or Cu, Au and In).
4. Void coefficient.

D. Alternate Coolants – measure buckling, spectral indices and activation traverses under room temperature conditions for D₂O and HB-40 coolants.

E. Verify buckling for reference lattice (room temperature) in PDP by substitution technique.

*All lattices employ reference lattice pitch and cluster micropitch and SE facility unless specified otherwise.

†Definition of Symbols

B^2 = Material buckling.

Lu-176/Cu = Activation ratio of Lu-176 to Cu capture in the fuel

ρ^{28} = Epithermal-to-thermal capture ratio in U-238.

δ^{28} = Total U-238 fissions-to-total U-235 fissions.

δ^{25} = Epithermal-to-thermal fission ratio in U-235.

$\text{Pu-239}_{\text{fission}}/\text{U-235}_{\text{fission}}$ = Activation ratio of Pu-139 to U-235 fission in fuel.

CR = Activation ratio of U-238 captures to U-235 fissions in fuel.

Activation flux traverses = Spatial activation measurements of Au, In, and a 1/v material such as Cu or Mn.

III. NEXT REPORT PERIOD ACTIVITIES

Effort will continue on the planning and analysis of critical experiments. HAMMER will continue to be modified especially through burnup capabilities to provide a better design tool. The application of collision probability methods to evaluate and possibly improve design packages will be studied further. Cross-section revisions considering the forthcoming ENDF-B evaluation will be initiated.

✓

✓

✓

✓

✓

HWO CR QUARTERLY REPORT

TITLE: Demonstration Plant Engineering

WBS NO: 14000

REPORTING PERIOD: October through December 1966

Direct effort on WBS No. 14000 was suspended at the end of the preceding report period. However, WBS No. 14000 work in progress at that time, and applicable to the Reference Plant as well as to the Demonstration Plant, was continued and is reported under WBS No. 15000.

1

2

3

4

HWO CR QUARTERLY REPORT	
TITLE: Reference Plant Engineering	WBS NO: 15000
REPORTING PERIOD: October through December 1966	

I. PROJECT OBJECTIVES

The overall objective of this activity is the design and construction of a 1000-Mwe HWO CR plant so that subsequent operation of this plant will provide:

- 1) Successful demonstration of economic operation of the HWO CR concept in a plant of this size.
- 2) Confirmation that the essential HWO CR features demonstrated by the 500-Mwe plant may be scaled up successfully to larger plants.
- 3) Cost and design data which will be used in defining large dual-purpose power and desalination plants.

The identification of areas requiring research and development effort is an additional objective.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. GENERAL STUDIES

1. Safety Studies

a. Loss-of-Coolant Accident in Containment Building

The mathematical equations describing the hydrodynamics, heat transfer, and thermodynamics portions of the "inlet header rupture" accident model were scaled and programmed into the analog computer. Preliminary computer runs were made to study the output from the hydrodynamics program with respect to operation of the main coolant system during and after loss-of-coolant accidents. Analysis of these runs indicated that for a large (24 to 48 in.) rupture in a Demonstration Plant reactor inlet header, the loss of system pressure would result in coolant flashing in the reactor outlet regions of the coolant system. Further study of the consequences of this type of accident, utilizing the entire model, is currently in progress.

b. Safety Aspects of Main Coolant Loop Operation

Several closed loop dynamic analyses were performed on the analog computer (see Section C. 4. a for a discussion of the model). The primary purposes of these analyses were:

- 1) Evaluation of main coolant loop isolation valve performance.
- 2) Determination of the system transient effects resulting from loss of coolant flow and from failure of a component.

The model included a representation of the nuclear, thermal, mechanical, and hydraulic characteristics of the primary system.

In evaluating isolation valve performance, certain requirements were established to prevent:

- 1) Fission product release from the containment building.
- 2) Excessive water injection into the reactor as the result of a steam generator tube failure.
- 3) Reverse flow in a loop.

Results of the isolation valve analysis appear in Figure 15000-1, which shows

2-6-67 UNC

66-7680-149-2

Figure 15000-1. Pump Turbine System Inertia vs Required Isolation Valve Closure Time (Gate Valves)

the effects of pump and drive turbine inertia on the valve closure time requirements for gate-type isolation valves. The two curves indicate the valve closing time requirements to prevent fission product release from the containment building, with and without the delay effect of a slug flow reactor outlet mixing tank. The two curves approach vertical asymptotes at about 22 and 9 sec, respectively, reflecting the case in which the pump does not shut down (infinite inertia). The horizontal asymptotes reflect the case of valve non-closure (flow stopped by system friction). The required closure time values will be somewhere between the two curves and will be dependent upon the amount of mixing permitted by the design of the reactor outlet mixing tank.

The closing time requirements of the valves to meet the fission product release criterion also proved sufficient to prevent water leakage to the core, provided the steam generator tube failure can be detected and valve closing action initiated within several seconds after the tube failure occurs. Because of stringent nonleakage requirements and pressure drop considerations, gate valves are preferable to butterfly valves for this application.

The closed loop system transients analog analyses showed that, for a shaft shear between the pump and drive of one main coolant pump, reverse flow in the absence of a check valve in the loop occurs in about 1-1/2 sec. The flow reduction through the core after this accident, without isolation valve closure, is shown in Figure 15000-2. With isolation valve closure, return coolant from the steam generator does not reach the reactor in time to result in significant inlet temperature changes.

Figure 15000-3 shows a reactor inlet temperature ramp of up to about 3°F/sec starting approximately 20 sec after a steam generator shell rupture accident in which all the water is assumed to be lost instantaneously. The severity of this ramp indicates a possible need for a high temperature scram on the main coolant system cold leg. Analysis of an accident in which loss of feedwater flow to one steam generator is experienced showed normal reactor control action to be sufficient to keep the resultant inlet temperature ramp in check.

Evaluation of the results of the isolation valve and system transient analyses showed that:

Figure 15000-2. Pump Drive Shaft Shear Reactor Coolant Flowrate vs Time

Figure 15000-3. Steam Generator Shell Rupture Reactor Coolant Inlet Temperature vs Time

1) An isolation valve closure time of between 9 and 20 sec, depending on the design of the reactor outlet mixing tank, is required to prevent fission product release from the containment building via the coolant loops.

2) A check valve is necessary in each coolant loop for,

- a) Prevention of reverse flow in the event of pump shaft failure
- b) Control considerations

3) Check valves must be restrained from closing quickly, preferably by snubbing action rather than by continuous damping.

4) The reactor cannot be maintained on-line after loss of coolant in one loop with the reference pump inertia, reactivity insertion rate, and magnitude of prompt positive coolant temperature coefficient of reactivity.

5) The loss of feedwater flow in a single loop does not pose excessive control or safety problems.

c. Tritium Hazards Associated with Moderator Cooling

Three proposed methods of moderator cooling are:

1) Direct river water system (heat rejection directly to a river water cooled heat exchanger)

2) Indirect river water system (heat rejection to a river water cooled heat exchanger via an intermediate fluid and heat exchanger)

3) Boiler feedwater system (heat rejection to the low-temperature feed-water heater)

Each of these methods was examined for potential tritium release to the public assuming a 10 gpm leak of D_2O into the cooling water. It was found that MPC in the cooling water is reached within one minute. This is the limiting case since no dilution outside the system was assumed. The direct and boiler feedwater systems require leak detection and automatic isolation action to prevent tritium release to the public. Leak detection and prompt manual isolation action was sufficient for the indirect cooling system. However, such action is predicated on the knowledge of a leak from the intermediate fluid to the river water. Under these conditions, a detection system for leakage from the intermediate fluid to the river water would also be warranted.

2. Fuel Element Dynamic Behavior

a. Axial Node Investigation

The single node and 5 axial node models were compared under static and dynamic conditions for the purpose of surveying the error in the single node

feedback reactivity. Five axial nodes were found to be quite adequate for simulating the transfer of heat to the moving HWOCR coolant for all conditions except for coolant inlet temperature perturbations which have frequency components greater than 7.5 rad/sec.

In steady state, the single node feedback reactivity is, (1) equivalent to that of the 5-node model for all power levels when coolant flow and coolant inlet temperature are held at design values, (2) more positive than that of the 5-node model when the coolant inlet temperature is larger than the design value at 100% flow and power, and (3) less positive than that of the 5-node model when the coolant flow is less than the design value at 100% power and coolant inlet temperature.

Dynamically, the transient feedback response to a rising power is more positive in the 5-node model than in the single-node model.

b. Radial Node Investigation

A study was made to determine the minimum number of radial nodes required in the uranium carbide fuel pin to simulate the HWOCR transients accurately.* The single radial node model for the HWOCR fuel pin was adequate for conducting scoping type transient analysis. To obtain more accurate representation of the fuel behavior, particularly during fast power transients, a two radial node model should be used. The use of more than two radial nodes has very little effect on the transient analysis results.

c. Results of Dynamics Studies

The response of the reactor to reactivity, coolant inlet temperature, and coolant flow perturbations is described elsewhere.† A small negative power coefficient tended to stabilize small fluctuations which result in reactivity changes; however, within the range of this study (up to 200% power), this

*C. R. Musick, "An Investigation of the Minimum Number of Radial Nodes Required in the Fuel Pin for Dynamic Analysis of the 750 Mwe CD-1 HWOCR Reactor," AI-CE Engineering Document No. TI-010-22-605, dated November 15, 1966

†C. R. Musick, "Results of the Five Axial Node Short Term Transient Analysis of the 750 Mwe CD-1 HWOCR," AI-CE Engineering Document No. TI-010-22-608

parameter had relatively little effect on the transient behavior of the reactor in response to large perturbations.

A comparison was made of results obtained from the three models (single-node analog, 5-axial-node analog, and 5-axial-node FORE-A) which have been used to simulate the dynamics of the 750 Mwe CD-1 HWOCR. These results showed that the analog and FORE-A 5-axial-node models generate power traces that agree to within 2% or better for all conditions that have been simulated. The single-node results are unconservative for reactivity and coolant flow perturbations but very conservative for coolant inlet temperature changes.

B. REACTOR DESIGN

1. Natural Uranium-Carbide Study

a. Thermal-Hydraulics

An optimization study of a natural uranium-carbide (UC) HWOCR reactor was completed this quarter, using the 1000 Mwe plant^{*} as a basis. The purpose of the study was to:

- 1) Determine the optimum set of design parameters for a natural UC reactor.
- 2) Aid in determination of the future potential of an HWOCR natural UC reactor.

Total power costs of potential 1000 Mwe designs were compared using equilibrium fuel cycle costs and 1975 economic ground rules with private financing. The complete set of ground rules for the fuel element, plant, and thermal-hydraulics are given in the report for the previous quarter.[†]

The reactor parameters which were varied in the optimization study are:

Number of Fuel Rods	19, 37
Coolant Outlet Temperature (°F)	650, 700, 750
D/U Ratio	24, 30, 36, 40
Number of Fins	3, 8, 12
Fuel Area (in. ²)	8, 6, 4
Fin Height (in.)	0.020, 0.030, 0.040

^{*}"HWOCR 1000 Mwe Nuclear Power Plant Preliminary Conceptual Design," AI-Ce-Memo-6 (October 1, 1965)

[†]"Heavy Water Organic Cooled Reactor Quarterly Technical Progress Report July through September 1966 AI-CE-55

NOTE: THE ABOVE CURVES DO NOT INCLUDE
OPERATION, MAINTENANCE AND LIABILITY

2-6-67 UNC

7680-25937

Figure 15000-4. Natural Uranium Studies (19-Rod Design)

Many designs at 650°F outlet temperature were eliminated because there was inadequate temperature difference between the organic coolant leaving the steam generator and the secondary steam temperature. For those cases where 650°F outlet temperature could not be used, outlet temperatures of 750, 725 and 700°F were used instead of 750, 700 and 650°F.

Selected curves showing the effect of variations in fin height, number of fins, D/U, and core exit temperature, for the 19-rod design are shown in Figure 15000-4. They are typical for both 19- and 37-rod designs. Lower costs are obtained with a lower exit temperature and fin height, higher D/U, and greater number of fins. In all cases the 4 in.² fuel area yields higher costs.

Based on minimum cost, the two cores which are optimum are compared in Table 15000-1. Although the required number of pressure tubes is somewhat less for the 37-rod design, the total power generating costs are less for the 19-rod design, principally because it has lower fuel cycle costs. This occurs because the burnup is greater and the fuel fabrication costs are less. Due to the lower overall cost, the 19-rod design was chosen as optimum for a natural UC HWO CR.

The study also indicated that the reactor is competitive with other nuclear power sources using 1975 ground rules, but does have a cost disadvantage of approximately 0.2 $\frac{\text{mill}}{\text{kw/hr}}$ compared with the enriched HWO CR design at an ore price of \$8/lb U₃O₈ and 83% parity.

b. Physics

Fuel cycle and basic axial power distribution data were evaluated over the range of parameters listed above. Additional variables in the physics analyses include the number of pressure tubes and pressure tube thickness. Calculations were also carried out for typical values of the above variables to provide estimates of the following factors:

- 1) Reactivity effects associated with gross radial power shaping by differential exposure zoning and reflector thickness,
- 2) Local axial peaking factor and reactivity effects (fresh fuel condition) associated with fuel bundle interface regions,

TABLE 15000-1
NATURAL UC FUEL ASSEMBLY DESIGNS

Element Design	19-Rod	37-Rod
Fuel area (in. ²)	6	8
No. of Fins	12	8
Fin Height (in.)	0.030	0.030
D/U Ratio	36	36
Power-to-coolant (Mw)	3083	3088
Core Exit Temperature (°F)	700	700
Core ΔT (°F)	170	164
Core Flow Rate (10 ⁶ lb/hr)	108	109.4
ΔP Fuel (psi)	125	130
ΔP Header (psi)	229	234
q' Maximum (kw/ft)	25.4	21.8
No. of Pressure Tubes	927	711
Volume Fraction Organic	0.301	0.306
Pressure Tube ID (in.)	3.747	4.335
Pressure Tube Thickness (in.)	0.072	0.083
Burn-up (Mwd/Mtu)	7400	6800
Pu-Content (g/kgU)	2.57	2.60
Reactor Capitalized Cost (mill/kwh)	0.3830	0.3251
Steam Plant Capitalized Cost (mill/kwh)	1.7172	1.7185
D ₂ O Cost (mill/kwh)	0.2762	0.2816
Organic Make-up (mill/kwh)	0.0363	0.0365
Fuel Fabrication Cost (\$/kgU)	20.3	22.5
Reprocessing Cost (\$/kgU)	16.3	15.75
Burn-up and Losses (\$/kgU)	-4.2054	-4.898
Shipping Costs (\$/kgU)	1.44	1.6697
Fixed Charges (\$/kgU)	5.449	5.281
Interest Charges (\$/kgU)	1.196	1.4031
Total Fuel Cost (mill/kwh)	0.7448	0.8364
Operation, Maintenance Liability (mill/kwhr)	0.30	0.30
Total Generation Cost (mill/kwhr)	3.457	3.498

3) Change in local radial power distribution with fuel burnup, i. e. distance along the fuel channel, and

4) Bias in discharge burnup predicted by a survey physics model.

The reflector thickness of 2 ft was selected for the cores fueled with UC of natural enrichment. This selection was based on the improvement in basic radial power distribution rather than any substantial reactivity increase resulting from the increasing of the reflector thickness from 1 to 2 ft. The maximum basic radial peaking factor (core center) was reduced by 10 to 15% and the peaking factor for the peripheral pressure tubes was increased by approximately 50%. The reactivity increase was of the order of 2 mk for a 700 pressure tube core; this reactivity gain is primarily due to an increased weighting of the lower exposure outer burnup zone and does not represent any significant increase in the core averaged discharge exposure.

Reactivity and power peaking effects associated with fuel bundle interface regions were evaluated with detailed R-Z cell calculations, and local radial power distributions along the fuel channel were defined from the results of one-dimensional (radial) cell depletion calculations.

The physics model employed to evaluate the average discharge burnup consisted of a lattice cell spectral calculation for the fresh fuel, a point burnup calculation using beginning-of-life cross sections, and an axial synthesis calculation to define the discharge burnup of the equilibrium core. In low enrichment cores having an initial conversion ratio greater than approximately 0.5, the build-in of plutonium isotopes leads to a significant variation in the cell averaged cross sections, with burnup and reduction in the discharge burnup predicted by the model for a given average feed enrichment. Calculations were carried out using middle-of-life cell averaged cross sections to estimate a bias in predicted burnup as a function of D/U ratio.

The fuel cycle data in Table 15000-1 is based on initial survey evaluations for the two cores. Upon selection of the core employing the 19-rod fuel assemblies, more detailed calculations were carried out to define the power distributions and fuel cycle data for the core. Table 15000-2 summarizes the physics data for the natural UC fueled HWOCR employing the 19-rod cluster fuel assembly.

TABLE 15000-2
SUMMARY OF PHYSICS DATA⁽¹⁾ FOR NATURAL UC FUELED HWOCR
WITH 19-ROD CLUSTER FUEL ASSEMBLY

Lattice Data	
Lattice pitch (in.)	11.0
UC fuel area (in. ²) ⁽²⁾	6.0
D ₂ O/UC area Ratio ⁽²⁾	17.46
Organic/UC area ratio ⁽²⁾	0.489
D/U (atom ratio)	36.0
Power Peaking Factors	
Basic radial ⁽³⁾	1.20
Basic axial ⁽⁴⁾	1.67
Local radial (fresh fuel condition)	1.13
Local axial (fresh fuel condition)	1.17
Excess Reactivity for Hot, Operating Conditions (equilibrium xenon and samarium)	5 mk
Fuel Inventory and Burnup	
Uranium totals (MTU)	232.63
Feed enrichment	Natural
Discharge exposure (Mwd/kgU) ⁽⁵⁾	7.830
Discharge U ²³⁵ (wt %) ⁽⁵⁾	0.220
Discharge fissile plutonium (gm/kgU _{initial}) ⁽⁵⁾	2.75

(1) For equilibrium core unless otherwise indicated.

(2) UC fuel smeared over radial gas gap; effective UC density of 12.94 g/cm³.

(3) Integrated channel output versus radial position for middle-of-cycle condition in all channels.

(4) Beginning-of-cycle condition in fuel channel near center of core.

(5) Core averaged value

2. 1000 Mwe Reactor Design — Fuel Handling and Storage

Possible advantages of locating the spent fuel storage pool outside the containment were investigated. While this arrangement might reduce the containment diameter, it was rejected because of possible release of radioactive material from defective irradiated fuel assemblies and because of the more complex containment penetration details required for the maintenance of coolant flow to fuel assemblies during their transfer through the containment.

A study was made of the requirements for installing and removing refueling machine heads to and from their trolleys for normal maintenance and during emergency conditions that require remote handling. Preliminary conceptual designs of equipment to perform these operations were prepared. The new and spent fuel transfer shuttles were relocated to a position midway between the upper and lower refueling machine maintenance rooms. During normal operation the maintenance room doors remained open so that the refueling machines could have ready access to the fuel transfer ports. This revision simplifies the refueling machine supply cable handling arrangement and permits removal of the complete carriage system into the maintenance rooms for maintenance and repairs.

C. PLANT ENGINEERING

1. Main Heat Transfer System — Steam Generators

Using the PESO II Program, a study was initiated to evaluate the effects of changing superheat approach and reheat steam temperatures and feedwater temperature on plant economics. Based on a steam generator cost of approximately \$20/ft² and a reactor outlet temperature of 750°F preliminary results indicated that the optimum superheated and reheat steam temperatures are lower than the 725°F temperature used in the Reference Plant Design. Varying feedwater temperature in the range of 390 to 450°F does not appear to change the plant economics significantly.

2. Coolant Recovery System

Possible methods of supplying hydrogen for use in the catalytic hydrocracking process of the Reference Plant were evaluated to determine the most economic

approach. Two methods were considered for recovery of H_2 from the hydrocracker and degasifier off-gas streams: (1) cryogenic separation of hydrocarbons from the H_2 , and (2) a diffusion method in which H_2 preferentially diffuses through a palladium barrier. Four methods of H_2 supply were evaluated: (1) electrolysis of water, (2) steam-hydrocarbon reforming, (3) dissociation of NH_3 , and (4) contracted H_2 supply by H_2 manufacturers.

With the exception of the use of a cryogenic H_2 recovery system, all the processes were shown to be technically feasible, and capital and operating cost data were obtained. Use of the cryogenic system was ruled out by equipment manufacturers because the water present in the waste gas stream must first be completely removed to prevent freezing problems in the cryogenic equipment and because the low temperatures required to enable separation of CH_4 from H_2 would make this process uneconomical.

This study showed that H_2 production by electrolysis, at \$0.11/100 SCF, is the most economical of the methods considered. The steam reforming method is the second most economical at an estimated \$0.16/100 SCF. For quantities required by the HWO CR (~4000 SCF/hr), steam reforming is usually the more economical method. However, because of the low anticipated power cost of HWO CR, the electrolytic method becomes the more economical system for this application. The power cost at which steam reforming becomes the more economical is approximately 6.0 mill/kw-hr. The electrolytic system is also preferred for simplicity of operation. Whereas the steam reforming system is a high temperature process (1300-1600°F) involving several processing steps, the electrolytic system operates as a single step process at essentially ambient temperature.

Based on the estimated operating cost of a palladium diffusion H_2 recovery system (\$0.21/100 SCF of H_2 recovered), there is no economic incentive to recover the H_2 from the off-gas system.

The results of this study are summarized in Table 15000-3 in terms of \$/100 SCF of H_2 .

TABLE 15000-3

H₂ COST (\$/100 SCF) FOR CATALYTIC HYDROCRACKER

H ₂ Supply Method	H ₂ Cost (\$/100 SCF)
Recover H ₂ by Pd Diffusion System	0.21
Electrolysis of Water	0.11
Steam-Hydrocarbon Reforming	0.16
NH ₃ Dissociation	0.26
Vendor-Supplied On-Site H ₂ Generator	0.25
Vendor-Supplied Liquid H ₂ Storage Tank; H ₂ Supplied on 10 year contract	0.42
Customer-Purchased Storage Tank; Liquid H ₂ Supplied by Manufacturer	0.39
Leased Storage Tank; Liquid H ₂ Supplied for Term of Lease	0.43

3. Turbine Plant – Turbine Bypass System

The behavior of the HWOCR during transient power operation may require the provision of a system that is capable of bypassing the entire plant main steam flow around the turbine to the condenser, on a continuous basis with the turbine-generator unit shutdown. To provide design and cost information for this possibility two alternate turbine bypass system concepts were prepared and evaluated. Concept 1 covers a system design which uses a portion of the bypass steam in all the extraction feedwater heaters to maintain the feedwater at normal operating temperature. The remaining steam is reduced in pressure and attemperated with condensate before it is bypassed to the condenser. Concept 2 differs from Concept 1 solely in that feedwater heating is done only in the de-aerator using a minimum amount of bypassed steam to maintain the deaerator pressure slightly above atmospheric pressure. Both designs are technically feasible and a review with equipment manufacturers did not indicate any significant design problems.

Concept 1 offers the advantage that the entire feedwater system is maintained at operating temperature and thermal shock to the steam generators is thus minimized during a turbine tripout.

Equipment and piping requirements were determined for each design and the costs associated with adding these systems to the Demonstration Plant were estimated. A summary of these costs is given in Table 15000-4. The estimated costs are \$423,000 and \$400,000 for Concepts 1 and 2, respectively. Since the cost differential is minimal, Concept 1, because of the advantage it offers, is preferred for the turbine bypass system design if reactor safety studies indicate definite requirement of a 100% turbine bypass system to maintain the full thermal load on the reactor after a turbine tripout.

TABLE 15000-4
SUMMARY OF ADDITIONAL DIRECT CAPITAL COSTS FOR
100% TURBINE BYPASS SYSTEMS

	Concept 1 (\$)	Concept 2 (\$)
Feedwater Heaters, Pumps, and Condensers	100,000	107,000
Piping and Valves	217,000	201,000
Control Valves and Desuper- heaters	106,000	92,000
Total Direct Costs	423,000	400,000

4. Plant Control System

a. Plant Simulation

The previously developed mathematical model of the HWOCR Demonstration Plant primary coolant system is shown schematically in Figure 15000-5. The model includes a representation of the nuclear, thermal, mechanical, and hydraulic characteristics of the primary system. The secondary or steam system inputs and outputs are shown to the left and right of the diagram. For the initial analyses, the secondary system inputs are assumed to remain constant or vary in a known manner with respect to time. Both loops are represented with provisions for the inclusion of check valves and various types of isolation valves. The major system components included are:

Figure 15000-5. Simplified Block Diagram of HWO CR Primary System

AI-CF-56
WBS 15000

2-6-67 UNC

7680-25938

- 1) Reactor
- 2) Reactor controller
- 3) Pumps
- 4) Pump turbine drivers
- 5) Evaporators
- 6) Superheaters
- 7) Reheaters
- 8) Isolation valves and actuators
- 9) Check valves
- 10) Piping, headers, manifolds, and mixing tanks

The mathematical model has been programmed for analog computer mechanization on four computer consoles using 280 amplifiers for the simulation.

b. Control Studies

Reduction and evaluation of computer study data pertaining to the controls analysis are currently in progress. Flow transients resulting from an emergency shutdown of one coolant loop are shown in Figures 15000-6 through -8.

Figure 15000-6. Reactor Coolant Flowrate Transients -
Loss of One Coolant Loop by Pump Tripout

Figure 15000-7. Reactor Coolant Flowrate Transients -
Loss of One Coolant Loop by Pump Trip
and Valve Closure

Figure 15000-8. Reactor Coolant Flow Transients -
Loss of One Coolant Loop by Pump Trip
or by Valve Closure

- 1) The coolant loop isolation valves possess gate valve flow characteristics.
- 2) Isolation valve closure time is 20 sec.
- 3) Check valves are installed on reactor inlet lines to prevent reverse flow.

Figure 15000-6 shows flow transients following trip-out of the pump in one coolant loop with various pump-turbine system rotational inertias. Reference pump-turbine inertia is approximately 37 lb-ft-sec^2 , and the steady-state reactor flow rate corresponding to one loop operation is approximately 65% of rated flow. Flow decay to steady-state single pump operation for the reference pump-turbine case would occur in just over 4 sec as shown on Figure 15000-6.

Curve 7 of Figure 15000-7 represents the flow transient resulting from an inadvertent closure of the isolation valve in one coolant loop. Curves 1 through 6 depict flow transients resulting from a pump trip-out with simultaneous closure of the loop isolation valves for the inertia conditions considered on Figure 15000-6.

Figure 15000-8 is a superposition of the 400, 600, and 1000 lb-ft-sec^2 inertia cases of Figures 15000-6 and -7 with and without isolation valve closure. Flow transients for pump-turbine inertias less than 400 lb-ft-sec^2 would not be affected by valve closure for the conditions specified, i. e., gate valve flow characteristics and 20-sec closure time.

One of the objectives of the computer studies was to determine the feasibility of maintaining "on-line" control of the plant during emergency shutdown of one coolant loop. A preliminary evaluation of these studies indicated the following:

- 1) The plant as designed in AI-CE-Memo-25* cannot be maintained on line during loss of flow in one coolant loop (with reference pump-turbine inertia). With the maximum reactivity rate limited to $0.005\% \Delta k/\text{sec}$, the control system cannot withdraw reactivity as rapidly as reactivity is inserted by the inherent positive feedback mechanism. The reactor increases in power level, exceeds protective systems setpoints, and scrams.
- 2) It is possible, though not necessarily desirable, to maintain the reactor on line during loss of flow in one loop if the reference reactivity withdrawal rate is increased and/or the reference pump-turbine inertia is increased.

*"Heavy Water Organic Cooled Reactor, 500 Mwe Nuclear Demonstration Plant Conceptual Design," AI-CE-Memo-25

Figure 15000-9 indicates the potential trade-off between inertia and reactivity rate, holding all other system parameters constant (i.e., valve characteristics, closure time, trip levels, trip delay times, etc.). The solid curve displays the trade-off relationship without isolation valve closure action. Control reactivity rate requirements, except for very large inertia systems, would exceed the present 0.005% Δk /sec; however, only negative reactivity (rod insertion) is required to initially set back reactor power level. Hence, compromise of present safety criteria would not be necessary.

Because of rapid flow decay, the isolation valve closure time of 20 sec is not a consideration for control reactivity rates, with the low inertia cases. However, when rotational inertia exceeds 400 lb-ft-sec², reactivity rate requirements increase due to the rapid increase in the rate-of-change of flow near the end of valve closure. The approximate curves of Figure 15000-9 were derived for feasibility studies and are applicable only for the specified conditions in protective system setpoints, delay time in initiating power setback, turbine speed in the operational loop, valve closure time, and valve flow characteristics.

Figure 15000-9. Pump-Turbine Inertia vs Reactivity Withdrawal Rate for Reactor Power Level Control in Event of Loss of Flow in One Coolant Loop

More than 200 computer runs were made to provide quantitative system dynamics data. These runs included the following investigations:

- 1) Emergency shutdown of one coolant loop resulting from,
 - a) inadvertent valve closure,
 - b) pump drive shaft severance,
 - c) pump drive seizure, and/or
 - d) pump-turbine tripout.

Parameter variations for these runs included valve closure time, isolation valve characteristics, pump-turbine inertia, control reactivity rate, and automatic power setback delay. These runs were made with and without check valves.

2) Coolant flow rate sensitivity studies for step, ramp, and oscillatory perturbations.

3) Reactivity step perturbation.

4) Reactor inlet temperature step and ramp perturbations (originating at the evaporator outlet).

5) Feedwater flow rate step and ramp perturbations at 100 and 60% rated power.

6) Reactor power setpoint ramp perturbations at 100 and 60% power.

7) Simultaneous feedwater flow rate and reactor power setpoint ramp perturbations at 100 and 60% rated power.

8) The effect of pump turbine throttle valve oscillations on pump speed and flow.

9) The effect of check valve damping and pump-turbine inertia on the initial pressure transient for both normal and "hung-up" check valve operation.

c. Steady-State Control Program

This part of the plant control mode evaluation, which consists of an analysis of the steady-state operation of the plant from 20% to maximum output, was continued using the following basic operating conditions.

- 1) Constant reactor coolant flow rate.
- 2) Variable reactor coolant average temperature, in accordance with Figure 15000-10.
- 3) Constant steam pressure at the turbine throttle, 915 psia.
- 4) Variable steam temperature to the turbine, according to the reactor temperature schedule and the steam generator heat transfer characteristics.
- 5) Constant condenser pressure, 1.5 in. HgA.

Figure 15000-10. Reactor Coolant Temperature Program

ance data obtained as a result of the investigations completed to date are presented below.

(1) Plant Efficiency Variation

Figure 15000-11 shows the variation in net plant efficiency as a function of the turbine operating temperature and percent of main turbine flow. The estimated curve for operation of the plant, based on the chosen control mode, is shown crossing the constant temperature lines. A comparison of this curve with that for a 725°F constant temperature control program shows an appreciably higher efficiency for the latter. These data can be used in the future as bases for economic evaluations of alternate control modes and other plant design features.

(2) Net Plant Output and Main Turbine Flow

Figure 15000-12 shows the equivalence of net plant output and percent main turbine flow at the operating temperatures under consideration. The percent main turbine flow has been used throughout as a convenient common parameter

*op. cit., AI-CE-55

Figure 15000-11. Net Plant Efficiency

2-6-67 UNC

7680-25944

Figure 15000-12. Net Plant Output and Main Turbine Flow

2-6-67 UNC

7680-25945

to correlate the performance of the main system components. This curve is included for convenience in obtaining the net plant electrical output corresponding to a given set of operating conditions.

(3) Reactor Thermal Power

Figure 15000-13 shows the reactor thermal power as a function of the turbine inlet steam flow and operating steam temperatures.

(4) Steam Generator

Figures 15000-14 through 15000-21 show the thermal requirements and required operating conditions of evaporator, superheater, and reheater as functions of the turbine inlet steam flow and steam temperatures.

The operating requirements shown in the foregoing figures, and the steam generator design data obtained from the feasibility study, are being used to obtain the performance of the evaporator, superheater, and reheater when operating under the proposed reactor coolant temperature program shown in Figure 15000-10 (constant reactor coolant flow, constant turbine inlet pressure).

2-6-67 UNC

7680-25946

Figure 15000-13. Reactor Fission Power Coolant Thermal Power

2-6-67 UNC

7680-25947

Figure 15000-14. Evaporator Thermal Power, Preheat Section

2-6-67 UNC

7680-25948

Figure 15000-15. Evaporator Thermal Power, Boiling Section

2-6-67 UNC

7680-25949

Figure 15000-16. Evaporator Thermal Power, Preheat and Boiling Sections

2-7-67 UNC

7680-25950

Figure 15000-17. Evaporator Outlet Steam Pressure and Temperature

2-7-67 UNC

7680-25951

Figure 15000-18. Feedwater Temperature

2-7-67 UNC

7680-25952

Figure 15000-19. Superheater Thermal Power

2-6-67 UNC

7680-25953

Figure 15000-20. Reheater Thermal Power

2-6-67 UNC

7680-25954

Figure 15000-21. Reheater Temperature Requirements

The steady-state control characteristics of the steam generator components, possible modifications of the temperature program, and the partial load operating conditions of the plant will result from this evaluation.

Evaluations to date indicate the presence of moisture in the steam flowing from the high pressure section of the turbine to the reheater when operating below 75% main steam flow. At approximately 40% steam flow, the moisture content is approximately 3%. Design features at the reheater inlet may be necessary to avoid erosion damage.

5. Utilization of D₂O Heat for Feedwater Heating

Additional work was performed to evaluate the effects of condenser pressure and turbine last stage bucket length on the net plant efficiency gain resulting from utilization of the D₂O heat for feedwater heating. Figure 15000-22 shows the differential improvement in net plant efficiency as a function of the condenser pressure and last stage bucket length. A condenser pressure range of 1.5 to 3 in. HgA was considered to be most likely, based on probable available cooling water temperatures. The improvement in net plant efficiency was evaluated on the basis of a tandem-compound 1800 rpm, 6-flow, turbine-generator producing a gross output of 1088 Mwe and operating with 900 psig, 725/725°F steam. The effect of the turbine last stage bucket length was evaluated for 38, 43, and 52-in. designs, which are the most probable for HWO CR plants.

Figure 15000-22. 1000 Mwe HWO CR Plant Improvement in Net Plant Efficiency Due to Utilization of D₂O for Feedwater Heating as a Function of Condenser Pressure

Figure 15000-23. Reactor Plant Heat Balance, 1000-Mwe HWOCR Plant, Natural Uranium Core (UC Fuel)

TABLE 15000-5

MAIN HEAT TRANSFER SYSTEM DESIGN DATA 1000 Mwe
NATURAL UC FUELED HWO CR

Number of Parallel Coolant Loops	3
Thermal Power to Coolant (Mwt)	3083
Total Reactor Coolant Flow (lb/hr)	108×10^6
Reactor Inlet Coolant Temperature (°F)	530
Reactor Outlet Coolant Temperature (°F)	700
Reactor Coolant Average Temperature (°F)	615
Coolant Temperature Rise Across Reactor (°F)	170
Pressure at Reactor Inlet Header (psia)	301
Total System Pressure Drop (psi)	301
Reactor Pressure Drop (header-to-header) (psi)	201
Main Loop Coolant Velocity (ft/sec)	25 maximum
Total Steam Generation Rate (lb/hr)	9.44×10^6
Reactor Coolant Volume (gal)	120,000 (includes mixing tanks)
Surge Tank Coolant Volume (gal) (0.8 full)	16,000
Piping Coolant Volume (gal)	100,000
Steam Generation Equipment Coolant Volume (gal)	70,000
Auxiliaries Coolant Volume (gal)	64,000
Total Coolant Inventory (gal)	370,000
Number of Main Coolant Pumps	3 (one per loop)
Flow Rate Each Pump (gpm)	78,600
Total Pump Head (ft)	760
Coolant Temperature at Pump (°F)	530
Shaft Horsepower Each Pump (hp)	16,000 (steam turbine drive)
Total Superheater Thermal Duty (Mwt)	367
Total Evaporator Thermal Duty (Mwt)	2333
Total Reheater Thermal Duty (Mwt)	405
Total Superheater Surface Area (ft ²)	87,000
Total Evaporator Surface Area (ft ²)	187,000
Total Reheater Surface Area (ft ²)	128,000

Figure 15000-24. Steam and Feedwater System Heat Balance, 1000 Mwe HWO CR Plant, Natural UC Core

TABLE 15000-6
TURBINE PLANT DESIGN DATA 1000 Mwe
NATURAL UC FUELED HWO CR

Reactor thermal output (including moderator and shield) (Mwt)	3,268
Reactor thermal power to coolant (Mwt)	3,083
Evaporator and superheater thermal power (Mwt)	2,700
Reheater thermal power (Mwt)	405
Gross generator electrical output (0.85 pf) (Mwe)	1,076
Plant auxiliary electrical power (Mwe)	12.0
Plant net electrical output (Mwe)	1,064
Plant net heat rate (Btu/kwh)	10,482
Plant net thermal efficiency (%)	32.6
Turbine arrangement	TC6F-52 in. LSB
Throttle pressure (psig)	600
Throttle temperature (°F)	675
Throttle enthalpy (Btu/lb)	1333.7
Throttle flow (lb/hr)	9.42×10^6
Reheat pressure (psia)	116
Reheat temperature (°F)	675
Reheat enthalpy (Btu/lb)	1365
Reheat flow (lb/hr)	8.5×10^6
Condenser pressure (in. Hg)	1.5
Turbine exhaust flow (lb/hr) (total)	7.05×10^6
Turbine Exhaust enthalpy (Btu/lb)	1022.4
Duty (Btu/hr) (total)	678×10^9
Generator rating (kva)	1,280,000
Generator terminal voltage (kv)	24
Generator power factor	0.85
Generator hydrogen pressure (psig)	60
Feedwater temperature (°F) (final)	385
Feedwater enthalpy (Btu/lb)	359.4
Shaft-driven auxiliaries	None
Steam-driven auxiliaries - hot reheat	Main feedwater and main coolant pumps

6. Natural Uranium Reactor Plant Design

The conceptual plant design for the 1000-Mwe natural UC fueled HWOCR optimization study (see Section 15000, B-1) was completed. The main coolant system design is shown diagrammatically in the reactor plant heat balance in Figure 15000-23; the design data are shown in Table 15000-5. The system contains three coolant loops and is similar to the design shown in AI-CE-Memo-6,* except that the main coolant pumps are located in the cold leg of the system.

The turbine plant arrangement is also similar to the design used in AI-CE-Memo-6, consisting of a TC6F-52 turbine with reheat, four extraction feed-water heaters, and moderator heat recovery in the feedwater. The plant has a net electrical output of 1064 Mw based on 600 psig, 675°F turbine throttle conditions and reheat steam at 675°F. The turbine cycle heat balance is shown in Figure 15000-24 and design data are summarized in Table 15000-6.

III. NEXT REPORT PERIOD ACTIVITIES

Preparation of system design descriptions for the principal plant segments will begin next quarter.

An optimized natural uranium 1000-Mwe HWOCR using uranium metal fuel will be developed.

Work on the analysis and design of the dump system for reactor vessel pressure relief will be completed.

Design reports for the reactor vessel and pressure tube assembly will be prepared.

A trade study report will be issued on the selection of materials for the reactor vessel.

A specification for the reactor vessel will be prepared.

*op. cit., AI-CE-Memo 6

HWO CR QUARTERLY REPORT

TITLE: Dual Purpose Plant Engineering

WBS NO: 16000

REPORTING PERIOD: October through December 1966

I. PROJECT OBJECTIVES

The dual-purpose (electric power production and water desalting) plant differs from the power-only plant in three major respects. First, the dual-purpose plant can utilize low temperature, high entropy heat in the evaporator sections of the plant, instead of in the low-pressure sections of the turbine. Second, in many areas of the country the requirements for water and power may become such that very large dual-purpose reactors of the order of 8,000 to 10,000 Mwt will be needed. The heavy water pressure tube reactor lends itself to extrapolation to these large sizes in single units. Third, the financing associated with water production or distribution plants differs from that associated with the investor-owned utility power plants. For investor-owned utility power plants, the fixed charge rates range from 10 to 14%. The fixed charge rate for water plants ranges from 4 to 7% in the large public water systems. This difference in fixed charge rates means that for the desalination of water, a high capital cost-low energy cost plant appears more advantageous than a high energy cost-low capital cost unit.

In the light of the above considerations, the objectives of this activity are as follows:

- a) Determine the feasibility of a 10,000-Mwt HWO CR plant.
- b) Determine the near-optimum parameters of fuel systems composed of UC, UO_2 , and U-metal. Determine the economic advantages of each of these systems at its optimum operating parameters.
- c) Determine optimum coupling schemes for a HWO CR type reactor for each of two water-to-power ratios for the optimum fuel system as determined in Paragraph (b).
- d) Perform conceptual designs of an optimized 3,000 Mwt and of a 10,000 Mwt dual-purpose HWO CR plant.
- e) Define areas and show incentives for additional research and development for desalting applications.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. PHASE B – FUEL ELEMENT OPTIMIZATION

1. General

Phase B of the Dual Purpose effort is about 80% complete. The maximum fuel temperature limits have been changed slightly. The revised values are shown below.

<u>Fuel</u>	<u>Old Value (°F)</u>	<u>Present Value</u>
UC	2200	30 kw/ft
UO ₂	4500	18 kw/ft
U metal	850	950°F

For the gap sizes required between the fuel and cladding, the temperature and linear heat ratings shown above for UC fuel, are nearly equivalent. The values were changed so that the parameter study could be run without concern for gas gap size and conductance, fuel conductivity, etc. The UO₂ peak temperature was changed because the 4500°F limit was too high when it was determined that cored pellets were required for the UO₂ fuel. The maximum fuel temperature limit of the U-metal was revised because a review of the data indicated that the severity of fuel growth is not increased if the temperature is increased from 850 to 950°F.

Many of the fuel systems in the Dual Purpose Study have been eliminated from further consideration. Those which were eliminated are discussed in the previous two quarterly progress reports.* Results from analysis of the remaining fuel systems are discussed in Sections B, C, and D below. The analyses were made of SAP clad UC rods, Zr clad UO₂ rods, Zr clad U metal rings, Al clad U metal rings, and Zr clad UO₂ inverse clusters.

The physics analyses required in support of the Dual Purpose Task B fuel element system design evaluation outlined in a previous progress report†

*"HWOCR Quarterly Technical Progress Report, April through June, 1966," AI-CE-54, and "HWOCR Quarterly Technical Progress Report, July through September, 1966," AI-CE-55

†Ibid, AI-CE-54

employs the parameters summarized in Table 16000-1 for the three principal fuel systems (UC-SAP rod clusters, UO_2 -Zr rod clusters, and U-metal annular elements). Two phases are shown in the table. The first phase explores the variation in nuclear properties of the equilibrium core, e.g., power distribution and fuel cycle data, with cross sectional area of fuel per channel, volume fraction of organic within the fuel channel, coolant exit temperature, and pressure drop across the core; the latter two quantities are not independent variables as far as the physics analyses are concerned but they are included in so far as they affect coolant temperature, pressure tube thickness, and number of fuel channels. Values for the average discharge fuel burnup and D/U atom ratio differ between the three fuel systems. The D/U ratio has been changed from that implied by the D/U parameter given in the guidelines.* This change

TABLE 16000-1
PARAMETERS EMPLOYED IN PHYSICS ANALYSES OF
THREE PRINCIPAL FUEL SYSTEMS

	Fuel System		
	UC-SAP Rod Clusters	UO_2 -Zr Rod Clusters	U-Metal Annular Elements
<u>Phase I*</u>			
Average Discharge Burnup (Mwd/kg U)	15	18	4, 6, 8
D/U	30	40	30
Fuel Area (in.^2)	4, 6, 8	4, 6, 8	3, 4, 5, 6
Volume Fraction of Organic	0.32, 0.39, 0.48	0.32, 0.39, 0.48	0.28, 0.33, 0.38
<u>Phase II†</u>			
Average Discharge Burnup (Mwd/kg U)	12, 15, 20	15, 18, 22	4, 6, 8
D/U	24, 30, 36	30, 40, 50	24, 30, 36

*Exit coolant temperature and pressure drop are additional variables (see text).

†For optimum fuel channel parameters defined in Phase I.

*Ibid, AI-CE-54

was influenced by three factors: (1) differences in uranium density for the three fuel materials, (2) lower bounds on lattice pitch dimensions imposed by mechanical limitations in the coolant manifold region, and (3) past experience on fuel cycle parameters.

The second phase of the analysis employs the optimum fuel channel defined during the initial phase for each fuel system. Fuel cycle and power distribution data are evaluated as a function of average fuel discharge burnup and D/U ratio. A third phase of the analyses, not shown in Table 16000-1, consists of a more detailed examination of a single core for each fuel system. The objectives are to verify fuel cycle and power distribution data for the equilibrium core; evaluate fuel and coolant temperature coefficients and coolant void coefficients to define trends between different fuel systems; and plan the fuel cycle and management data for the initial core.

The following two sections are relevant to the fuel element systems in general; details pertinent to either the rodged or annular fuel systems are described in subsequent sections.

a. Fuel Cycle Data

Fuel cycle data for the equilibrium cores consist of feed fuel enrichment and fissile plutonium and U^{235} content of the discharged fuel. The feed fuel enrichment is selected to give 5 mk excess reactivity in the equilibrium core; reactivity allowances for fuel bundle-end effects, and differential exposure zoning to shape the basic radial power distribution, are included.

b. Power Distributions

Basic radial and local axial peaking factors are defined on the basis of past analyses and/or calculations for typical parameters where required to meet conditions peculiar to designs being investigated herein. Limiting local axial peaking factors are taken to be 1.15 and 1.10 for the rodged and annular fuel element designs, respectively. A 1-ft radial reflector and basic radial peaking factor of 1.16 are employed for the cores with rodged fuel element designs, whereas a 2-ft reflector and basic radial of 1.20 are associated with annular element cores.

Basic axial power distributions for the beginning, middle, and end-of-cycle conditions within the average fuel channel are evaluated by axial synthesis of the bidirectional equilibrium core. The basic axial power distribution within a channel is primarily a function of the discharge exposure; second order effects are fuel enrichment and D/U ratio. The axial power shape is symmetrical, e.g., approximately a cosine, in the discharge exposure range of 6 to 10 Mwd/kg U; below 6 Mwd/kg U the peak is shifted slightly towards the channel exit direction due to the initial rise in k_{∞} with plutonium build-in, and above 10 Mwd/kg U the power distribution exhibits a roof-topping effect which increases with discharge exposure. The basic axial peaking factor goes through a maximum (1.70-1.75) for channels having axial power distributions symmetrical about the midplane of the core and decreases to approximately 1.50 for the roof-topped power distributions in channels discharging at approximately 20 Mwd/kg U. Differences in integrated channel power output between beginning and end-of-cycle channel conditions are a function of the channel discharge exposure. For channels discharging below 8 to 10 Mwd/kg U the channel power output at the end-of-cycle condition is greater than the beginning-of-cycle condition, whereas for channels discharging at higher exposures the reverse is true. The disparity in channel power output between beginning and end-of-cycle conditions shows a progressive increase as the discharge exposure deviates above or below 8 to 10 Mwd/kg U.

The local radial power distributions for fuel assemblies not employing radial enrichment zoning, i.e., rod-type fuel assemblies, exhibit the trends illustrated in previous preliminary conceptual design reports. The beginning-of-life local radial peaking factor for the 37-rod clusters is dependent on fuel area and enrichment; D/U ratio and volume fraction of organic have a negligible effect on the local radial peaking factor for fresh fuel over the range investigated.

Several parametric cost curves had previously been developed showing the variation in cost of the reactor primary heat transfer loop (in ¢/million Btu of reactor fission power) with the reactor ΔT , reactor ΔP , and various turbine plant steam conditions. A discussion of these cost curves appeared in the two preceding quarterly reports.* In these studies, the primary heat transfer loop cost was based on the following items.

*Ibid, AI-CE-54, AI-CE-55

TABLE 16000-2
PRIMARY HEAT TRANSFER LOOP STUDY CASES

Case Number	Reactor Outlet Temperature (°F)	Steam Temperature (°F)	Steam Pressure at TT (psig)	Reactor ΔT Range (°F)
1	750	725	900	100 to 200
2	750	725	600	100 to 250
3	750	725	400	100 to 290
4	650	625	900	60 to 100
5	650	625	600	60 to 140
6	650	625	400	80 to 180
7	550	525	400	45 to 90
8	500	475	165	45 to 105
9	700	675	900	60 to 140
10	700	675	600	100 to 200
11	700	675	400	100 to 220
12	725	700	900	100 to 180
13	725	700	600	100 to 220
14	725	700	400	100 to 260
15	675	650	900	85 to 130
16	675	650	600	90 to 170
17	675	650	400	100 to 200
18	625	600	600	60 to 120
19	625	600	400	80 to 160
20	600	575	600	40 to 100
21	600	575	400	80 to 140

- 1) Organic coolant makeup (includes credit for hydrocracking coolant)
- 2) Organic coolant inventory
- 3) Primary loop equipment; viz., steam generators, piping, valves and fittings, main coolant pumps and drives
- 4) Auxiliary organic processing equipment
- 5) Indirect capital costs.

During the report period, additional primary loop cost curves for a broader range of reactor temperatures were developed. A summary of all of the cases studied is tabulated in Table 16000-2. The unit cost for each case was determined for two reactor pressure drops, 150 and 250 psi. Fixed charges for the reactor plant were based on 12% per year over the plant life for private ownership and 7% per year for public ownership. The fixed charges include amortization, taxes, insurance (except for nuclear insurance), and interim replacements. For the organic inventory charges, rates of 10 and 5% per year were used for private and public ownership, respectively. A plant factor of 90% was assumed. Typical cost curves for the primary heat transfer loop are shown in Figure 16000-1.

2. Rodded Fuel Designs

a. Thermal Hydraulics

Analyses and evaluations of the rodded fuel systems are essentially completed. This effort includes variations of fuel area, organic volume fraction, reactor coolant exit temperature, and economic ground rules. The primary emphasis was given to evaluation based on 1985 public ownership ground rules, average discharge exposures of 15 Mwd/kg U (UC) or 18 Mwd/kg U (UO₂), and a D/U of 30 (UC) or 40 (UO₂). With these parameters as a basis the most promising designs with respect to costs for UC and UO₂ fuel were chosen for more detailed investigation of average discharge exposure and D/U ratio. The basic parameters for the most promising designs are as listed below.

	<u>UC</u>	<u>UO₂</u>
Total fission power (Mw)	3192	3155
Fuel area per pressure tube (in. ²)	8	8
Organic volume fraction	0.326	0.318
Reactor coolant exit temperature (°F)	725	700
Reactor coolant ΔT (°F)	185	114
Number of pressure tubes	561	737
Fuel assembly pressure drop (psi)	168	168

Figure 16000-1. Primary Loop Cost vs Reactor ΔT

The thermal-hydraulic portion of the study is to establish the basic reactor parameters as a function of the variables investigated and to establish the basic cost evaluation parameters. Each design is restricted by operational limits established for the study. A list of these restrictions for the UC-SAP and UO_2 -Zr systems follows.

	<u>UC</u>	<u>UO_2</u>
Maximum heat release (kw/ft)	30	18
Maximum cladding temperature ($^{\circ}\text{F}$)	850	800
Maximum pressure drop across fuel strings (friction + entrance and exit) (psi)	160	160

While maintaining these limits, variations of organic volume fraction (0.3 to 0.48), fuel cross sectional area (4, 6, and 8 in.²), and reactor coolant exit temperatures (600 to 700 or 750 $^{\circ}\text{F}$) were investigated.

An increase in the organic volume of a design effectively increases the hydraulic diameter. Since the designs are pressure-drop limited, increasing the hydraulic diameter permits higher velocities and therefore higher heat transfer coefficients. Designs with higher organic volume fractions, that are surface temperature limited, have fewer required pressure tubes and higher power densities. This is especially true for the UC systems where an increase in fin height (i.e., an increase in heat transfer area) accompanies an increase in organic volume fraction.

As reactor coolant exit temperature is reduced, higher power densities (fewer pressure tubes) can be achieved until the maximum heat release (kw/ft) limit is reached. If the coolant temperature is lowered beyond this limit, the required number of pressure tubes remains constant and the velocity can either be decreased or kept constant with a lowering of the surface temperature. In this survey the velocity was reduced to take advantage of the lower pumping power and thinner pressure tube.

Higher fuel areas result in fewer required fuel columns because the fuel element has a greater heat transfer area and hydraulic diameter than with the lower fuel area. Higher heat transfer coefficients result from higher hydraulic diameters for a given pressure drop. These factors offset the increased local radial peak to average power associated with the higher fuel areas.

In general, the UO_2 -Zr designs have more fuel channels and lower ΔT 's than the UC-SAP designs. This is primarily due to the lower heat transfer area, kw/ft limit, and allowable surface temperature.

b. Physics

The rodged fuel element designs consist of the UC and UO_2 fueled 37-rod cluster assemblies. By the close of this report period both phases of the analysis (see Table 16000-1) were completed for the UC-SAP rod clusters and Phase II was nearing completion for the UO_2 -Zr rod clusters.

Fuel cycle data evaluated during the initial phase of the analysis exhibit the following trends. The feed enrichments required for equilibrium cores fueled with UO_2 -Zr rod clusters are approximately 0.2 wt % higher than UC-SAP fueled cores employing comparable fuel areas and volume fractions of organic. For the range of variables investigated, changes in volume fraction of organic resulted in the largest variations in feed enrichment (approximately 0.15 to 0.22 wt %). Variations in feed enrichment with fuel area were small and dependent upon the magnitude of the volume fraction of organic. For example, the largest variation occurred at a volume fraction of organic of 0.47 where the feed enrichment for the cores fueled with 4 in.² UC per channel increased by approximately 0.05 wt % over that required for the nominal 6 and 8 in.² UC per channel. The fissile plutonium content of fuel discharged from the UC cores was 3 to 5% higher than the more highly moderated UO_2 cores. Small changes in fuel cycle data were observed for variations in coolant exit temperature due to compensating changes in the number of pressure tubes and pressure tube thickness, i.e., an increase in pressure tube thickness usually accompanies an increase in number of pressure tubes with the consequence that the equilibrium core eigenvalue is essentially unchanged. Similar compensating effects reduced the variation in feed enrichment with fuel area, viz., the cores having small fuel areas generally have a larger number of pressure tubes and a smaller pressure tube thickness than cores having larger fuel areas.

Variations in fuel cycle data with changes in D/U and discharge burnup for the UC cores were as follows. Feed enrichment increased by only a few percent as the D/U ratio decreased from 36 to 24 but increased at a rate of 3 to 4% per Mwd/kg U as the discharge burnup increased from 12 to 20 Mwd/kg U. The

fissile plutonium content increased by approximately 10% as the D/U ratio changed from 36 to 24 and as the burnup increased from 12 to 20 Mwd/kg U.

c. Cost Evaluation

In coupling the reactor to a steam plant, consideration must be given to the reactor coolant exit temperature and ΔT . For each variation studied the reactor was coupled to a plant with the highest steam pressure possible for the operating parameters. Due to the higher efficiencies associated with higher steam pressures, this combination produced the lowest total costs.

The costs for these systems were based on a power only system and exclude the capital costs of the turbine generator complex. The cost unit is defined as the cost of the steam from the steam generator in ¢/million Btu divided by a typical net turbine cycle efficiency for those steam conditions.

The reactor capital costs for these systems are closely related to the number of fuel channels and therefore to reactor size. Thus the reactor capital cost decreases with lower reactor exit temperatures and higher organic volume fractions.

The secondary plant costs are related to the steam condition governed by reactor coolant exit temperatures and ΔT 's. In general lower costs are realized at higher exit temperatures due to higher possible steam pressure and greater turbine cycle efficiencies.

The fuel cycle costs are greatly influenced by the fuel equilibrium feed enrichment. The enrichments are dependent on various parameters; however, the major contributors to the fuel cycle costs are organic volume fraction, efficiency, and pressure tube thickness. These costs are lower for smaller organic volume fractions and higher exit temperatures except in cases where the inlet temperature is increased sufficiently to cause an increase in the pressure tube thickness.

For lower fuel area, the number of pressure tubes is more sensitive to organic volume fractions and thus a greater spread in reactor capital costs exists than at higher fuel areas. This variation produces a reverse trend in the total costs as a function of organic volume fraction. The higher volume fractions give lower total costs at low fuel areas with the reverse being true with high fuel areas. This trend is also influential in producing lower total

costs for the higher fuel areas. The trend in reactor capital cost, and the lower fuel cycle costs at the lower organic volume fractions, provide minimum cost of the high fuel area, low organic volume fraction designs.

The UC-SAP designs have lower total costs than the UO_2 -Zr designs. This is primarily due to the higher fuel enrichment requirements and the greater number of pressure tubes.

3. U-Metal Concentric Ring Designs

a. Thermal-Hydraulics

Analysis of the concentric ring designs with Zr and Al finned cladding for the three sets of economic ground rules and average discharge exposures of 4, 6, and 8 Mwd/kg U have been completed. Evaluation of the effect of variations of D/U is in progress.

During this report period the design approach for concentric ring elements was changed from uniform fuel enrichment in each fuel ring to variable enrichment. This new approach results in a flattening of the local radial power distribution and a subsequent 30% reduction in the peak-to-average discharge exposure. This is especially important to exposure-limited fuels such as uranium metal. With variable enrichment, the geometry for each design is established by equal coolant ΔT 's in each subchannel and nearly equal fuel thickness criteria. This geometry will produce nearly equal surface and centerline temperatures in each fuel ring, exclusive of the engineering hot channel factors, thereby permitting the highest power density design.

The most promising designs, with respect to total costs, for the Zr clad and the Al finned clad systems have been selected for the additional analysis required to determine the optimum D/U variation and calculate the levelized annual cost. The basic parameters for these designs are listed below.

	<u>Zr Clad</u>	<u>Al Clad</u>
Total fission power (Mw)	3320	3324
Fuel area per pressure tube (in. ²)	6.0	6.0
Organic volume fraction	0.281	0.337
Reactor coolant exit temperature (°F)	700	685
Reactor coolant ΔT (°F)	139	155
Number of pressure tubes	1094	742
Fuel assembly pressure drop	160	160

The above results established the parameters necessary for the cost evaluation. These values are governed by mechanical and metallurgical limitations established for the survey. The major limitations are listed below.

- 1) Maximum cladding surface temperature of 800°F for Zr clad designs and 750°F for Al clad designs
- 2) Maximum U-metal fuel temperature of 950°F
- 3) Maximum fuel column pressure drop of 160 psi.

Three design parameters were initially varied to establish trends. These parameters were fuel area, organic volume fraction, and reactor coolant exit temperature. The following trends were determined from the analysis:

- 1) Reducing reactor coolant exit temperature results in increased power densities with an associated reduction in number of pressure tubes and increased reactor ΔT . This is primarily due to the fact that the surface temperatures and the fuel temperatures attain their respective limits simultaneously.
- 2) Increasing the organic volume fraction or fuel area also increases the power density. This develops from the increased heat transfer area and the increased velocity associated with the increased hydraulic diameter. In addition higher organic volume fraction designs have smaller fuel thicknesses.

Comparing the results for designs with Zr cladding to those with finned Al, the Al designs in general have fewer pressure tubes and higher coolant ΔT 's. This is primarily due to the increased heat transfer area of the fins which more than compensates for the decreased allowable surface temperature and smaller hydraulic diameters. The increased ΔT 's limit the range of exit temperatures investigated due to the inlet temperature being lower than saturation temperature of the steam conditions that are economically attractive to the ground rules of this study.

b. Physics

Preliminary physics data were evaluated for two types of Zircaloy clad annular metal elements. The first employed nearly equal fuel areas and the same U^{235}/U^{238} ratio in each of three concentric fuel annuli; the second had a nearly uniform fuel thickness in each of three annuli and a variation in the U^{235}/U^{238} ratio between annuli. The principal difference between the two fuel elements is the peak-to-average integrated burnup of the fuel in a typical plane of the fuel element, i.e., away from bundle-end effects. For the element with the variable

U^{235}/U^{238} ratio the peak-to-average burnup is within a few percent of unity whereas for the other element the peak-to-average and maximum-to-minimum burnup ratios are approximately 1.3 and 1.75, respectively. The disadvantages of the zoned fuel element are threefold:

- 1) Average U^{235} content of the fresh fuel bundle required to achieve a given k_{∞} is approximately 10% higher than its counterpart due to preferential placement of U^{235} near the center of the element where both the U^{235} fission cross section and thermal flux are lower.
- 2) Inner ring feed enrichments are 1 wt % or greater and the U^{235} content of discharged fuel for this ring is well above diffusion plant tails.
- 3) Local to bundle average power density (including local axial effect) is maximum in the inner ring for the fresh feed condition; in elements not employing radially varying U^{235} distributions the variation in local radial is greater than the increase in the local axial with decreasing radial position, but in the zoned element the local radial decreases with increasing radius for fresh fuel condition.

Since the U-metal elements are burnup-limited, the improvement in peak-to-average burnup and reduction in the local radial peaking factor for the zoned fuel element were taken to be sufficient grounds for carrying out all subsequent analyses with this fuel element design.

In addition to the Zircaloy clad element, analyses are being carried out for cores employing annular metal elements clad with finned aluminum plus a nickel diffusion barrier. At the close of this report period, the initial phase of the analysis was complete for both fuel systems. The average feed enrichment required for the finned Al-Ni clad elements is approximately 0.2 wt % higher than for the Zircaloy clad elements. This increase is primarily due to the higher parasitic capture rate per fuel channel for a given fuel area and volume fraction of organic. The major contributor is the Al-Ni cladding. The additional pressure tube thickness and organic associated with the larger fuel channel are second order effects.

The ratio of wt % U^{235} in the inner to outer ring varies from 2 to 3 depending on fuel area, volume fraction of organic and discharge exposure. No significant difference in this ratio is observed between the two types of fuel elements, thus the increased enrichment required for the Al-Ni clad elements is distributed proportionately between the three fuel annuli.

c. Cost Evaluation

The calculation of the cost components for these systems was performed in a similar manner as for the rodged designs. Each design was coupled to a secondary system with the highest possible steam pressure (400, 600, or 900 psi) as governed by the coolant exit temperature and ΔT .

The trends associated with the costs for the systems are similar to those for the rodged designs. The reactor capital cost decreases with lower coolant exit temperature due to the fewer required pressure tubes. Lower plant capital costs exist at the higher exit temperatures due to the higher efficiencies. Because of the lower enrichments, minimum fuel cycle costs are associated with lower organic volume fractions.

At the lower fuel area designs the minimum organic volume fraction is limited by the coolant ΔT for a given coolant exit temperature. This is especially true for the finned Al clad designs, where the flow is reduced with the smaller hydraulic diameters associated with the fins.

Thus, as with the rodged designs, lower total costs are associated with high fuel areas, low organic volume fractions, and high coolant exit temperatures. The Zr clad designs have lower total costs than the Al clad designs, due to the higher fuel enrichments required by the additional parasitic material in the element, i.e., Al, Ni, and organic.

4. Zr Clad UO_2 Inverse Cluster

A partial parameter study was made to determine the performance of a Zr clad UO_2 inverse cluster.

The inverse cluster fuel assembly which was chosen for this study fits into a 4.280 in. ID Zircaloy pressure tube and has one flow hole size. With more than one allowable flow hole size in a given fuel element, the number of inverse cluster configurations which can fit in a pressure tube of a given size is practically unlimited. With one hole size, the best geometries are obtained with either 19, 37, or 61 holes. Sixty-one holes was not considered because of the small hole size (1/4 in.) and minimum web thickness (<0.1 in.) which is required.

The 37-hole configuration is superior to the 19 because, for the same pressure tube size, it has higher fuel area, lower organic volume fraction, and higher power removal capability. Therefore, detailed analysis was performed on the 37-hole configuration.

The inverse cluster consists of 37 0.375 in. ID Zircaloy tubes and is contained in a 4.28 in. ID shroud. The thickness of the Zircaloy tubing was 0.020 in. The tubing is free-standing, but the outer shroud which is 0.040 in. thick collapses on the OD of the fuel in order to limit its thickness. (A minimum wall thickness greater than 0.100 in. would be required if it were free-standing.)

The fuel could be either sintered UO_2 discs with 37 holes or it could be vibratory-packed. If sintered fuel is used, it will be necessary to maintain good dimensional control on the pellets. A 0.040-in. thick cylindrical shroud encases the UO_2 and 37 flow tubes. The end plates are attached at each end by TIG welding to the shroud and flow tubes. The inverted cluster is 4.08 in. in diameter. The flow gap between the ID of the pressure tube and the OD of the bundle is 0.020 in.

Analysis was performed using the ground rules for the Dual Purpose Study. A limited parameter study was conducted on the 37-hole inverse cluster design to determine the required number of pressure tubes to remove 3000 Mwt, as a function of average reactor outlet temperature. A comparison of it and a typical 37-rod* SAP-clad UC fuel element indicates that two or more times as many pressure tubes are required for the inverse cluster design, primarily because the heat release from the inverse cluster is very nonuniform. Although the inverse cluster has a fuel area of 7.56 in.², compared to 6 in.² for the UC design, the weight per unit length of each fuel system is about the same (34.1 lb/ft). There are several serious complications associated with the design of an inverse cluster:

- 1) Since the fuel assembly is one welded unit, it would be extremely difficult to accommodate relative thermal expansion between flow tubes and across the whole fuel assembly.

*Since analysis has not been completed on all SAP clad UC designs, a typical design having a pressure tube ID near 4.28 in. was chosen. The 37 SAP-UC fuel element design has 6 in.² of fuel, a rod OD of 0.543 in., and an organic volume fraction of 0.391 (0.063 in. fin height) and fits into a 4.26 in. ID pressure tube.

- 2) The outer shroud would have to collapse on the fuel in order to limit the thickness of the outer shroud.
- 3) Zircaloy hydriding could be a serious problem, particularly in the fuel supported shroud.
- 4) Relative changes in gas gap conductance during operation could adversely affect the temperature distribution in the fuel assembly.
- 5) There is no irradiation experience which would indicate what burnup might be achievable in an inverse cluster.
- 6) Fabrication of fuel assemblies having sintered UO_2 fuel would be more expensive than SAP clad UC rods and a theoretical density of not more than 85% could be expected if vibratory packed fuel were used.

Since the Zircaloy clad UO_2 inverse cluster design offers no apparent advantage and, in fact, has many disadvantages compared to SAP-UC, it does not appear to have future potential for HWO CR Dual Purpose use.

B. PHASE C - COUPLING STUDIES, MODERATOR HEAT USAGE STUDY

This study, completed during the report period, was made to compare the relative economics of either recovery or rejection of the heat generated in the D_2O moderator of a 3000-Mwt dual-purpose water desalting plant. The procedure used in the study was discussed in the previous quarterly report.* A summary of the cases evaluated is presented below.

Method of Moderator Heat Removal	Turbine Back-Pressure	
	1.51 psia	35 psia
Reject to ocean (direct and indirect)	1†	3
Preheat boiler feedwater (direct and indirect)	2	4
Recover in brine topping heater of water plant (direct and indirect)		5
Recover (by vapor compression cycle) in brine topping heater (indirect only)		6
Recover in inlet brine preheater (direct and indirect)		7

The "indirect" methods utilized an intermediate fluid in a closed circuit between the D_2O and heat sink. Figure 16000-2 shows the relative water costs in cents per 1000 gal. for each case listed.

*Op. cit., AI-CE-55

†These are case numbers referenced in Figure 16000-3.

2-6-67 UNC

7680-25958

Figure 16000-2. Relative Water Costs — Indirect and Direct Cooling System

Case 6 utilized the heat pump principle. This method removes D_2O heat at a low temperature and transfers it to the heat sink at a higher temperature. A typical schematic diagram of this cycle is shown in Figure 16000-3. Moderator heat is transferred to the working fluid in the evaporator of the vapor compression system. The working fluid is then compressed and the temperature is increased by the heat of compression. Transfer of heat to the water plant occurs in the heat pump cycle condenser.

For a noncondensing turbine back-pressure of 35 psia, relative water costs based on a moderator outlet temperature of $190^{\circ}F$ were determined and compared for the moderator heat dissipation methods listed below.

- a) Reject to ocean (direct and indirect)
- b) Recover (by vapor compression cycle) in brine topping heater (indirect only)
- c) Recover in inlet brine preheater (direct and indirect)

The relative water costs are indicated graphically in Figure 16000-2, Cases 3, 6, and 7. The relative water costs for Cases 3 and 7 are the same and, for both of these cases, are less than for Case 6. In Cases 3 and 7, the direct cooling method is slightly lower in cost than indirect cooling. However, the safety considerations of direct cooling must be evaluated before the final design selection is made.

The previous quarterly report discussed the fact that two of the proposed heat recovery methods (Cases 4 and 5) require an increase in the moderator operating temperature to be considered for the 35 psia back-pressure plant. For instance, to be able to recover moderator heat in the brine topping heater of the water plant (Case 5), the moderator system must be designed to operate with an outlet temperature of $317^{\circ}F$ instead of the current reference design temperature of $190^{\circ}F$. To recover moderator heat in the boiler feedwater system of the 35 psia back-pressure plant (Case 4), the moderator outlet temperature must be increased to $333^{\circ}F$. It was also determined that an additional capital cost of up to approximately \$3,000,000 could be invested to make the moderator system compatible with the higher temperature before the resulting water cost became equivalent to the case where D_2O heat is rejected to the ocean.

2-6-67 UNC

7680-25959

Figure 16000-3. Typical Schematic Diagram of Heat Pump
Principle Cycle (Case 6)

A cost estimate was made to determine the additional costs associated with an increase in moderator operating temperature from 190 to 333°F. Operation at the higher temperature requires an increase in the design and operating pressure of the calandria tank in order to prevent bulk boiling of the moderator. To raise the design pressure of the calandria tank, the tubing wall and tank wall must be increased in thickness. The increase in calandria tube wall thickness increases parasitic absorption and, as a result, increases the fuel enrichment requirement or decreases the attainable discharge exposure. From the calculated effects on power generation costs, it was possible to obtain an equivalent capital cost due to the greater parasitic absorption cost increase. Combining this equivalent capital cost with the capital cost attributed to the thicker calandria tubing and tank wall material, a cost of \$3,900,000 was estimated for increasing the moderator operating temperature. Therefore, it was concluded that recovering moderator heat in the brine topping heater of the water plant or in the boiler feedwater system of the 35 psia back-pressure noncondensing turbine plant results in higher unit water costs than either rejecting the heat to the ocean or recovering it in the brine inlet preheater.

III. NEXT REPORT PERIOD ACTIVITIES

The effort being expended as part of Phase B will be completed next quarter. A fuel element system will be recommended for use in the conceptual designs required by Phase D. The economic and technological lines of reasoning for this selection will be promulgated in a "Phase B Final Report."

The moderator heat usage studies will be published as an engineering document and will be distributed.

Based upon the results of Phase B, conceptual designs of a 3000 Mwt and of a 10,000 Mwt dual purpose plants will be initiated. These designs will show how an HWOCR can be best utilized in Dual Purpose applications.

1

2

3

4

5

HWO CR QUARTERLY REPORT

TITLE: Dual Stream Reactor Concept

WBS NO: 16000A*

REPORTING PERIOD: October through December 1966

I. PROJECT OBJECTIVES

The dual stream reactor concept involves two physically separated coolant streams operating in parallel, both drawing heat from a single reactor and utilizing the energy to produce steam at two different temperatures. This concept was developed for application in a dual purpose plant — the high temperature stream to be used for electric power production as well as to produce process steam for desalination and the low temperature stream to be used for desalination only. The object of the present study is to determine the feasibility of a dual stream HWO CR and to determine its major features.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. NUCLEAR ANALYSIS METHODS

1. Batch Refueling

Batch refueling, rather than on-line refueling, is considered for the dual stream concept. A discharge fuel exposure of 20,000 Mwd/ton is used in the calculations. A recommended refueling is three batches per core loading at approximately annual intervals; this is simulated in the point reactor analysis by requiring criticality at an exposure of (2/3) of 20,000 Mwd/ton.

2. Criticality

Criticality is taken at $k_{\text{eff}} = 1.035$ with the 3.5% bias to account for maneuvering (1.5%), absorption by otherwise unaccounted for structural materials (1%) and additional leakage due to power shaping (1%).

*Work performed in part at Rensselaer Polytechnic Institute under subcontract CE-HW-2.

3. Nuclear Codes

HAMMER, THERMOS, and CRUMB codes are used in essentially the same fashion as in the single stream HWO CR.

4. Mid-Life Spectrum for Group Averaging

Initial burnup calculations used cross sections corresponding to beginning of life; reference enrichment and mid-life composition were selected on the basis of these calculations. Cross sections for mid-life composition and concomitant spectra were then computed and these spectra were used for cross section averaging from beginning to end of life in new burnup calculations; these calculations were used as the basis for lifetime predictions, reactivity coefficients and feed enrichments.

5. Reactivity Coefficients

Reactivity coefficients for fuel temperature, organic temperature and D_2O vapor fraction were computed at mid-life with cross sections and k_{eff} recomputed for perturbed conditions.

6. Range of Parameters

As in the thermal and hydraulic analysis described in the last progress report,* considerations covered fuel element sizes from 1 to 9 rings, (volume D_2O)/(volume UO_2) from 6 to 20. Initial enrichments considered ranged from 1 to 2 wt %.

B. NUCLEAR ANALYSIS RESULTS

1. Beginning of Life Reactivity

Table 16000A-1 gives a typical case of beginning of life reactivity results. The table gives k_{∞} and also gives the effective separate 4-factor formulation of k_{∞} and the non-leakage probability as k_{eff}/k_{∞} . The rather large and cancelling effects of the resonance escape probability and fast fission factor are shown explicitly. With respect to reactivity, it is clear from the table there is an optimum fuel element size and volume of D_2O to fuel ratio.

*op. cit., AI-CE-55

TABLE 16000A-1
BEGINNING OF LIFE REACTIVITY RESULTS FOR AN
ENRICHMENT OF 1.5 wt % U²³⁵

Number of Rings	$\frac{D_2O}{UO_2}$	ηf	p	ϵ	k_∞	P _{NL}	k_{eff}
1	6	1.5898	0.6433	1.1607	1.1871	0.9880	1.1728
1	10	1.5898	0.7477	1.1042	1.3126	0.9853	1.2933
1	15	1.5876	0.8139	1.0759	1.3902	0.9822	1.3655
1	20	1.5844	0.8512	1.0618	1.4320	0.9793	1.4023
3	6	1.5777	0.6680	1.1630	1.2257	0.9874	1.2102
3	10	1.5755	0.7589	1.1142	1.3322	0.9838	1.3106
3	15	1.5708	0.8148	1.0893	1.3942	0.9795	1.3656
3	20	1.5653	0.8453	1.0770	1.4250	0.9753	1.3898
6	6	1.5669	0.6721	1.1802	1.2428	0.9857	1.2251
6	10	1.5607	0.7416	1.1402	1.3196	0.9803	1.2937
6	15	1.5513	0.7798	1.1209	1.3560	0.9735	1.3201
6	20	1.5406	0.7990	1.1123	1.3691	0.9667	1.3235
9	6	1.5569	0.6582	1.2028	1.2326	0.9837	1.2126
9	10	1.5459	0.7114	1.1697	1.2865	0.9761	1.2558
9	15	1.5301	0.7400	1.1542	1.3069	0.9662	1.2627
9	20	1.5130	0.7548	1.1472	1.3101	0.9560	1.2525

The major features of the dual stream fuel element, and the effect on the nuclear behavior by the variation in size, as reflected by the number of rings, is shown in Table 16000A-1. With increasing D₂O/UO₂ volume ratio, the thermal neutron regeneration factor (ηf) decreases as expected due to the additional absorption of thermal neutrons by the hydrogen added in with the D₂O. For fixed D₂O/UO₂, ηf decreases with size due to the thermal flux depression within the fuel element, which becomes more severe as size is increased. Here added absorption by the hydrogen in the D₂O and the clad on the outside of the fuel element tend to reduce ηf .

The variation of resonance escape probability (p) with the D_2O to UO_2 ratio is not surprising. As D_2O is added, there is more efficient slowing down and a better probability of avoiding U^{238} resonances. At fixed D_2O/UO_2 , p shows a maximum as the size of the fuel element is increased. For the smaller fuel element, the major effect as size is increased is the self-shielding within the fuel element in the U^{238} resonance region. For very large fuel elements, the predominant effect is an increase in the flux in the fuel relative to the D_2O as fuel element size increases (with constant D_2O/UO_2 volume ratio). In effect, the more remote parts of the D_2O region are experiencing a lower and lower flux as they are further away from the fuel element. Therefore, the total number of collisions within the fuel element relative to collisions in the D_2O increases as size is increased, and U^{238} resonance collisions also increase, with the result being a lower p . The combination of these two effects produces a maximum in p with fuel element size.

2. Feed Enrichment

Results of burnup calculations are given in Table 16000A-2. Feed enrichments are selected such that $k_{eff} = 1.035$ at 13,330 Mwd/ton. The lower feed enrichments in the table reflect the lower need of enrichment to maintain criticality to 13,330 Mwd/ton of those cases which started out with the higher values of k_{eff} in Table 16000A-1. Another smaller but noteworthy effect is more absorption in U^{238} as shown by the lower p in Table 16000A-1, produces more Pu and hence a smaller rate of change of reactivity with burnup.

3. Effect of Mid-Life Cross Sections

Tables 16000A-1 and 16000A-2 are based on beginning-of-life cross sections. Repetition of burnup calculations with mid-life cross sections showed significant differences only for the lower values of (volume D_2O)/(volume UO_2) regardless of the fuel element size.

The dependence of the fast fission effect (ϵ) on D_2O/UO_2 and element size is shown in the table. This factor includes all fissions above thermal energies. ϵ decreases with an increase of D_2O/UO_2 due to the more efficient slowing down of neutrons as D_2O is added. ϵ increases with fuel element size because of the greater probability for collision of a high energy neutron with a U^{238} atom to produce a fast U^{238} fission in the larger fuel element.

TABLE 16000A-2
FUEL INVENTORY DATA FOR 20,000 Mwd/MTU EXPOSURE

Number of Rings	$\frac{D_2O}{UO_2}$	Feed Enrichment (wt %)	Discharge Enrichment (wt %)	Fissile Plutonium Content g/kgU
1	6	2.05	0.64	5.5
1	10	1.53	0.27	3.9
1	15	1.40	0.17	3.2
1	20	1.37	0.14	2.9
3	6	1.81	0.48	4.8
3	10	1.46	0.24	3.6
3	15	1.38	0.17	3.1
3	20	1.38	0.15	2.8
6	6	1.68	0.42	4.5
6	10	1.45	0.25	3.7
6	15	1.41	0.22	3.3
6	20	1.44	0.21	3.2
9	6	1.68	0.44	4.7
9	10	1.51	0.31	4.1
9	15	1.51	0.30	3.9
9	20	1.58	0.32	3.8

The non-leakage probability (P_{NL}) directly follows the variation in fast and thermal diffusion lengths since all cases shown in the table were computed with the same total buckling. In the cases studied, the age changed very little, and all changes in leakage are attributable to changes in thermal diffusion length. The variation with both D_2O/UO_2 and fuel element size is such that the greater the flux depression within the fuel element (i. e. the more collisions in D_2O relative to fuel), the larger the thermal diffusion length and hence the greater the leakage and the lower P_{NL} . This increase in the thermal diffusion length is brought about in the described fashion by either increasing the size of the fuel element or by adding D_2O .

Due to this shift in cross section averaging, changes of less than 8% in the feed enrichment and less than 13% in the Pu concentration occurred.

4. Reactivity Coefficients

Reactivity coefficients for selected cases are given in Table 16000A-3. As expected, fuel and D₂O coefficients are negative and the organic coefficient is

TABLE 16000A-3
REACTIVITY COEFFICIENTS

Number of Rings	$\frac{D_2O}{UO_2}$	D ₂ O Void (% Δk /% Void)	Organic Temperature (Δk /°F x 10 ⁻⁵)	Fuel Temperature (Δk /°F x 10 ⁻⁵)
1	6	-0.34	-2.9	-3.1
1	10	-0.25	+1.1	-1.3
1	15	-0.20	+1.8	-0.82
3	6	-0.27	+0.05	-1.6
3	10	-0.21	+2.2	-0.99
3	15	-0.16	+3.1	-0.60
6	6	-0.22	+1.1	-1.4
6	10	-0.14	+3.4	-0.90
6	15	-0.08	+4.3	-0.65

positive. Since, as the fuel element increases in diameter with (volume D₂O)/(volume UO₂) held constant, the flux in the fuel element rises relative to the D₂O at all energies and there are more collisions of all sorts within the fuel element. Hence, the effect of changes in organic and fuel temperature become larger with fuel element size and the effect of D₂O density becomes smaller. Increasing the D₂O volume while keeping the fuel element fixed decreases the effect of D₂O voids and increases the positive organic temperature effect since larger D₂O volume supplies more moderation, lessening the influence of organic moderation but not relative absorption. Also, the fuel temperature effect becomes smaller with this increasing D₂O because there is relatively more room for slowing down past the U²³⁸ fuel resonances and less Doppler effect.

C. FUEL UTILIZATION AND COST

1. Figures of Merit

Fuel utilization figures of merit have been applied to this fuel cycle. These are essentially the same as those quantities defined elsewhere.*

Calculation of fuel costs reflects changes in feed enrichment, burnup and Pu content only. Fabrication costs are expected to vary widely with size of dual stream element and it is not clear how they compare to those of present HWO CR. Since analysis of fabrication costs is considered beyond the scope of this present work, fabrication and reprocessing costs in \$/kgU were taken from a later report.† The same values were used for all cases.

2. Comparison with HWO CR

For comparison, fuel cycle data for the 500 Mwe Demonstration Plant§ and the 1000 Mwe Plant† were used. Equilibrium core fuel utilization and costs were computed using the same methods. The results given in Table 16000A-4 show that, apart from fabrication and reprocessing costs which have not been evaluated, several of the dual stream designs compare favorably to HWO CR.

D. EVALUATION OF WORK TO DATE

The effort on the Dual Stream HWO CR concept was completed during the quarter. A detailed report covering the entire effort will be issued shortly. The major conclusion of this study is that a Dual Stream HWO CR is feasible. The next step would be a detailed economic analysis of this concept. Some conclusions of the study are listed below.

- 1) A Dual Stream HWO CR is feasible. Further analysis is required to establish the economics of the concept. The design of the fuel element, a 2-pass multi-tube element with organic coolant passing through the tube appears feasible from the nuclear, thermal and hydraulic points of view.

*"Quarterly Technical Progress Report, Inception Through June 1965," AI-CE-3, p 0910-32

†op. cit. AI-CE-MEMO-6

§op. cit. AI-CE-MEMO-25

TABLE 16000A-4
FUEL UTILIZATION AND COST

Number of Rings	$\frac{D_2O}{UO_2}$	Fuel Cost* (¢/10 ⁶ Btu)	Static System Utilization (Mwdt/gm)	Dynamic Utilization (Mwt-yr/ton)
1	6	10.0	0.64	9.5
1	10	7.0	0.77	12.7
1	15	6.6	0.88	14.8
3	6	8.4	0.70	10.8
3	10	6.7	0.81	13.5
3	15	6.6	0.89	15.0
6	6	7.5	0.73	11.6
6	10	6.6	0.81	13.5
6	15	6.6	0.85	14.2
Demonstration Plant		6.8	0.85	14.8
1000 Mwe Plant		7.1	0.79	14.0

*Fabrication and reprocessing costs for the Dual Stream Fuel Element were arbitrarily made equal to those of the rodged type used in the HWO CR Reference and Demonstration Plants. (Fabrication costs = \$36/kgU and Reprocessing cost = \$20/kgU for all costs.)

2) The energy split between the organic and D₂O stream can be varied in the range of 16 to 46% of the heat going to the D₂O by varying the size of the fuel element and the number of organic coolant tubes.

3) Reactivity gained due to the lower organic and clad content in the dual stream elements is more than sufficient to compensate for reactivity losses due to D₂O boiling and reactivity required for burnup between refueling.

4) Due to the effects of lower organic content in the fuel element and to the boiling of the D₂O moderator, very slow reactivity transients result from perturbations in reactivity, organic inlet temperature, or flow.

5) In the event of loss of flow or loss of coolant accidents in the organic stream, adequate cooling of the fuel element by the D₂O stream can be accomplished only in the case of the one ring element.

HWO CR QUARTERLY REPORT	
TITLE: Pressure Tube	WBS NO: 21000
REPORTING PERIOD: October through December 1966	

I. PROJECT OBJECTIVES

The overall objective of this project is to develop a pressure tube assembly and associated components for use in the HWO CR. Included in the work is the testing of components and representative assemblies under simulated environmental conditons, to ensure and demonstrate that the goals of structural adequacy, reliability, maintainability and service lifetime can be met.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. SUMMARY

The major milestones for the report period were as follows.

- 1) Fabrication of TS 20-1A was completed; fabrication of 20-4 was 75% completed.
- 2) Fabrication of a SAP pressure tube for WR1 (EXP-WR1-2001) was initiated.
- 3) Phase I of TS 10-4 was completed, and Phase II was initiated. Phase II was prematurely stopped due to a failure in the lower SAP-to-SAP flash weld.
- 4) TF-20 loop construction was essentially completed and shakedown was initiated.

B. PRESSURE TUBE SYSTEM DEVELOPMENT

1. General

The pressure tube asembly, End Item Specification CPO14N21001, dated December 2, 1966, was reviewed. Appropriate test planning is in process which is consistent with development test requirements of the specification. The initial development tests to be performed in Test Facility 20 are summarized

TABLE 21000-1
TF-20 PRESSURE TUBE DEVELOPMENT TESTS

Test	Test Title	EI Specification Paragraph	Principle Investigations and Uncertainties	Test Article				
				20-1A	20-4	20-5	20-6	20-7
A	Pressure Tube & Fuel Element Vibration, Displacement, and Wear	4.1.1.5 4.1.1.6 4.1.1.7 4.1.2.2	Natural or induced vibration; vibration mode; structural fatigue; resonance; flow induced displacements; fretting damage to pressure tube; life expectancy; effect of flow direction		X			
B	Thermal Hydraulic Performance	4.1.1.1 4.1.1.2 4.1.1.3 4.1.2.2	Pressure drop, fuel elements aligned and misaligned; ΔP vs time; ΔP with internals; ΔP vs flow and temperature; heat losses; temperature profiles; local boiling in end shields or moderator; thermal shock, fatigue; distortion; leakage	X	X	X		
C	Corrosion, Erosion, Oxidation	4.1.2.2	Chemical and galvanic corrosion; surface oxidation of carbon steel components, effect of scaling on surfaces and components; effect of H ₂ O leakage in gas gap (20-1A only)	X	X	X	X	X
D	Structure Performance	4.1.1.4 4.1.2.2 4.1.2.4	Pressure cycling effect, thermal relaxation; elastic, inelastic deformation, failure modes; creep; effect of structural overloads; guide bearing performance; galling; friction; wear from refueling, fuel fin wear		X	X	X	X
E	Organic, CO ₂ Leakage	4.1.1.8 4.1.2.2 4.1.2.4	Leakage vs thermal and pressure cycles; leakage vs time; factors affecting leakage; performance of gaskets and seals; leak detection sensitivity, response, accuracy	X	X	X	X	X
F	Accelerated Life	4.1.2.1 4.1.2.2 4.1.3 4.1.4 4.1.5	Life expectancy; stress and temperature effects, design margin verification; life cycle operation				X	X
G	Off-Design Performance	4.1.2.2 4.1.3	Extreme conditions of temperature, pressure, flow; mechanical loading; failure modes and consequences; acceptable limits; gas gap overpressure effect		X		X	

in Table 21000-1. The design, manufacturing, liaison and pretest planning for the WR1 pressure tube irradiation test section (EXP-WR1-2001) were initiated (See Section II-B-4).

A pressure tube assembly reliability study was initiated. The components which are considered critical to the stable operation of the pressure tube assembly with consideration to overall plant availability are as follows.

- 1) SAP pressure tube
- 2) Bearings
- 3) Torque stops
- 4) Transition joints
- 5) Shield plug and latches
- 6) End seal and mechanism

The concept of margin testing (test to failure) is being evaluated as a solution to the demonstration of reliability. Margin testing of the components will basically consist of accelerated tests to failure and actual environmental tests to failure (where possible, in a reasonable time period). The best stresses at which to conduct these tests and the number of specimens to be tested at a given stress level are being determined by analysis.

2. Component Development

Status of the component development effort is as follows.

a. In-Core Section

Testing of the SAP 895, 4-in.-diameter in-core pressure tube sections continues as part of the effort in Test Facility 10. Testing of TS 10-4 (SAP tube with hot pressed joints and flash welds) was initiated and had completed 518 hr of test time when failure occurred at the lower transition joint flash weld. A structural analysis, including bending and torsional loading, was initiated to determine stress levels at time of failure. Operation will be resumed with an in-core section utilizing sandwich-type rolled transition joints (TS 104A).

Qualitative sliding wear tests of four materials on the inner surface of SAP 895 tubing were completed. The materials tested were 7 and 10% SAP, Aluminum, and 300 series stainless steel. The tests were performed in a

450 - 500°F organic bath. The samples were representative in geometry to the fuel bundle protruding surfaces sliding against the pressure tube inside surface, as shown in Figure 21000-1. The following observations were made.

2-10-67 UNC

7680-25994

Figure 21000-1. Laboratory Wear Test Setup

1) The first material combination tested was aluminum on SAP. Both the aluminum and the SAP wore appreciably. A great deal of pickup was observed on both materials.

2) A small amount of wear was observed using 304 stainless Steel on SAP 895.

3) In those cases where the contact pressure and reciprocating motion were the same for 7 and 10% SAP on SAP 895 (10%), less wear was noted for 7% SAP. Data scatter tended to mask any differences where test conditions were not duplicated.

A plot showing the relationship between contact pressure and the ratio of groove depth to total length of travel for the material combinations tested is shown in Figure 21000-2.

b. Shield Plugs

Due to problems encountered with cracking of the Homalite test housing during hydrostatic tests, flow model tests were not performed during this reporting period. New test housings will be made from Plexiglas, which will defer the flow visualization studies to next fiscal year.

Figure 21000-2. Wear Test - Various Specimens on SAP 895 Tubing

The carbon steel shield plugs (reference design AI-CE-25) were operated for 504 hr at nominally 600°F in TF-10. Pressure drop data were obtained at flow rates of 150 to 380 gpm, corresponding to a velocity range of 15 to 34 ft/sec in the TS 10-4 Phase I fuel bundles. Figure 21000-3 shows the delta P-flow test relationship. A discrepancy between experimental and calculated performance values indicates a need for flow visualization studies; however, from a mechanical standpoint, the shield plugs functioned satisfactorily. There was no evidence of galling or wear on the sliding surfaces. There were no obvious indications of fretting or surface damage which would be caused by vibration. A

0.1-in. stress relaxation was noted in the Inconel-X spring free length. Additional tests will be performed to determine if stress relaxation is a potential problem area.

The first set of mock-up shield plugs for TS-20 test sections (Figure 21000-4) were received. The items are constructed of carbon steel and are equipped with an Inconel - X spring. These shield plugs will be equipped with orifice plates so that the shield plug ΔP may be adjusted to appropriate values for pressure tube system test purposes.

c. Guide Bearings

A literature search was initiated on high-temperature sliding wear data for use in the selection and evaluation of sleeve-type guide bearings. Negotiations are in progress with AECL for the use of one of their wear testing machines to be used in bearing mock-up and component reliability tests. The capabilities of this machine are well suited for HWOCR application and are as follows.

Figure 21000-3. Shield Plug Pressure Drop

Figure 21000-4. TF-20 Shield Plug

Temperature: ambient to approximately 800°F

Environment: air, carbon dioxide, nitrogen or others (15 to 100 psia)

Stroke: 0.040 to 0.535 in. (can be easily modified for others)

Frequency: 80 cycle/hr to 1 cycle/day

Load: up to 3200 lb

Specimen Size: 3-3/4 to 8-in. diameter; (length not critical)

A testing laboratory was contacted for the purpose of performing small-scale material coupon tests. From data indicated in the literature, screening tests performed on coupons are expected to aid considerably in initial bearing material selections. Results of this testing are expected to be used for material qualification for the current developmental and WR1 pressure tube test sections.

d. End Fitting

From the results of testing the B. F. Goodrich single-flex radial seal, it is concluded that:

- 1) Seal leakage is dependent on the thermal transient rate of organic in the end seal region. Transient temperature rates in the end seal region in excess of 5°F/sec (from 700 - 400°F) tend to cause leakage rates in excess of 5 cc/hr.
- 2) After several hundred hours of operation, a permanent diameter distortion in excess of 0.003 in. occurs.
- 3) Seal leak rates of less than 0.5 cc/hr can be easily attained where static temperature in the end seal region is 400°F or less.
- 4) At 750°F, the leak rate is sporadic and varies from 0.5 cc/hr to nominally 10 cc/hr.
- 5) An uninsulated seal housing increases the seal effectiveness during a thermal transient.

A modified design of an end seal was completed. This modified design will be tested to determine if it meets the requirement of an organic leakage less than 0.04 cc/hr. The modified design, utilizing a positively-actuated Inconel 718 seal, offers the following advantages.

1) Pressure assist on the sealing face; i. e, the internal pressure is used to produce a greater sealing force. This feature is expected to make a tighter seal with less sensitivity to thermal transients.

2) A loading spring is not required in the mechanism. As a result, the inherent problems encountered with spring relaxation are eliminated and a more compact mechanism is obtained.

3) Positive radial retraction of the sealing lip. Permanent set, which caused some difficulty in the original B. F. Goodrich seal removal operations, is not expected. Nominal diametral clearance during seal retraction is 0.005 in.

A test fixture was designed for testing this seal. The fixture can be used either in the TF-50 test loop or attached to either end of the TS 20-5 pressure tube test section assembly.

e. Flow Vanes and Thermal Baffle

From past experiences, the pressure tube end seal is sensitive to higher operating temperatures and thermal transients. A thermal baffle-flow vane assembly was designed to reduce the metal temperature in the vicinity of the end seal. Figure 21000-5 shows the pressure tube end seal with thermal baffle assembly. A temperature distribution of this assembly was made following soaking for 5 hr in 550°F organic flowing at 300 gpm.

The temperature of the thermal baffle assembly is shown in Figure 21000-6. These data indicate that the baffle will reduce the temperature in the end seal region 250°F less than the bulk organic. Additional testing of this baffle with organic flow in the opposite direction is required. Flow visualization tests for the flow vanes and thermal baffle plate with transparent models will be performed, in the next fiscal year, to determine flow profiles and mixing.

3. Assembly Development

Fabrication of TS 20-1A (SAP 895 pressure tube with sandwich rolled joints; separate reactor vessel - shield tank) was completed and a trial assembly was made. Integral test instrumentation was designed and is being installed. This instrumentation will be used to obtain temperatures, pressure tube displacement,

7680-25998

Figure 21000-5. Pressure Tube End Seal With Thermal Baffle Assembly

test section pressure drop, moderator and shield tank heat losses, and CO_2 leak rates from static seals.

Fabrication of TS 20-4 (SAP 895 pressure tube with hot pressed joints; integral reactor vessel — shield tank) is approximately 75% complete. A delay of approximately one month was encountered due to the rejection of material for the thin-walled simulated calandria tube. Design of integral test instrumentation was initiated. Usage will be similar to that indicated for TS 20-1A.

Design of a third test section, TS 20-5, was completed. The design has provisions for reference design latching shield plugs and end fittings. Procurement action will be initiated during the next report period. Expected delivery is early in next fiscal year.

2-10-67 UNC

7680-25999

Figure 21000-6. Temperature Distribution of Thermal Baffle Assembly

C. SINGLE CHANNEL TEST LOOP (TF-10)

The following is a summary of the TF-10 activities during the reporting period.

- 1) Completed TS 10-4 Phase I test program.
- 2) Initiated TS 10-4 Phase II test program December 9, 1966. Due to failure of the bottom SAP-to-SAP flash weld on the pressure tube assembly, the TS 10-4 Phase II test program was stopped on December 10, 1966.
- 3) Removed the TS 10-4 Pressure Tube Assembly from the calandria tube and cut off the failed section of the pressure tube for a detailed metallographic evaluation of the weld joint.
- 4) Initiated fabrication of a new pressure tube assembly (TS 104A) incorporating Model 15 rolled transition joints.

Considering the obvious wear problem with the TS 10-2 configuration, the following changes were made in an attempt to determine methods of preventing wear during TS 10-4 testing.

- 1) Modified inlet configuration to reference design to determine if inlet hydraulic conditions were contributing to wear.
- 2) Pre-wore the TF-10 fuel bundle fins to provide more bearing surface on the pressure tube wall.
- 3) Installed wear bands and a center band on one TF-10 fuel bundle and one center band on another bundle to provide more bearing surface on the pressure tube wall and to prevent fuel bundle bowing.

The evaluations of some of the parameters being checked during the TS 10-4 Phase I program were as described below.

1. Pressure Tube Internal Wear

Visual examination of the TS 10-4 pressure tube after Phase I testing (500-hr operation) indicated internal wear on the pressure tube at all bundle locations, but with insignificant depth noted at the banded bundle locations 4 and 5. The results of the inspection are shown in Figure 21000-7. Also, it was noted that wear at bundle locations 1 through 3 very closely matched wear marks noted on the fuel bundle fins. At bundle locations 4 and 5, wear marks matched wear marks on the solid fuel rods; no fin wear was noted. Actual wear marks as viewed through a borescope are shown in Figure 21000-8. Considering the internal pressure tube wear insignificant at bundle locations 4 and 5, the following conclusions on pressure tube internal wear apply.

- a) Bands on bundles prevent wear at bundle locations 4 and 5. Bands would possibly prevent wear at locations 1, 2, and 3.

2-11-67 UNC 7680-251001

Figure 21000-8. TS 10-4 Wear
Marks on Pressure Tube ID
as Viewed Through
Borescope

b) The change to reference design inlet conditions and prewearing of fuel bundles did not prevent wear on unbanded bundles of the particular configuration tested.

A Scoping Test Plan was written for the next series of tests in TF-10 which are concerned primarily with wear of the pressure tube. The first tests will use a string of half-length fuel bundles. The second phase of the program will use several full-length fuel bundle designs.

The fabrication of ten 22-in. long dummy fuel bundles for tests in the TF-10 and TF-20 facilities was completed. The bundles are similar to the reference design bundles described in AI-CE-Memo 25 except that they are one-half the length. All radial dimensions were reduced slightly to permit the bundles to be installed in the 4-in. ID test facilities. The end plates are

the stainless steel stepped design, and the cladding material is 7% Al_2O_3 SAP. Stainless steel dummy pellets were used to fill the tubes.

Each assembled bundle was turned in a lathe to reduce the OD and to provide more bearing surface on the peripheral fins which contact the pressure tube. The assembled fuel bundles were cast in wax for the purpose of keeping the fuel pins and fuel bundle rigid.

Five of the bundles were banded at the mid-section by use of a 0.015-in. - thick by 1/2-in. -wide stainless steel band. The nominal clearance between fins of adjacent cladding tubes in the unbanded fuel assembly is 0.008 in. Except for one banded fuel assembly which has clearance between rods, the rods in the banded fuel assembly are pulled in so that the fins contact one another. Fins at the periphery of the fuel assembly, where the bands are located, were grooved to permit the band to be flush with the outside envelope of the bundle and to hold the band in place. The two ends of the band were joined by butt welding.

Five 44-in. -long bundles with flat end plates are currently being fabricated. Some of the bundles will be unbanded. One will be banded at the midplane and another will be banded at three axial, equally spaced locations.

It is planned that two more bundles with wider fins (0.050 in.) on the tubes will be fabricated. These bundles will use flat end plates and will be similar to the full-length reference design bundles. All of the full-length fuel bundles will contain magnets to measure lateral vibration of fuel rods.

2. Fuel Bundle Vibration

Vibration data taken during Phase I of the TS 10-4 test led to the following observations.

a) No correlation between the two variables in the test environment, temperature and flow, could be established due to the low magnitude of vibration detected and the lack of a sufficient number of repetitive test runs. In general, vibration in the order of 0.002 to 0.003-in. amplitude and 3 to 5-cps frequency was detected at most magnet locations at flows between 150 and 380 gpm. No vibration was detected at zero flow conditions.

b) Frequencies and amplitudes appeared to be randomly distributed and not a function of vertical position along the length of the string.

c) No vertical (axial) vibration was detected.

d) Vertical (axial) displacement as a function of flow was observed.

Figure 21000-9 shows the distribution of the vibration amplitudes along the string. A magnet (No. 17) was inserted near the top of the fourth fuel bundle for TS 10-4 Phase I tests in order to determine whether or not any axial fuel vibration was present. None was observed. If this observation holds for all of the fuel bundles, which seems likely, then axial movement of the fuel on the pressure tube would not be the mechanism causing pressure tube wear.

The axial displacement of the magnet due to flow introduces a question as to the reliability of the data obtained on the other magnets since their "gain" is sensitive to axial displacement. More accurate positioning at each flow rate will resolve this question. Figure 21000-10 shows the plot of axial displacement of magnet No. 17 vs flow rate for Phase I of the TS 10-4 test.

Testing similar to that conducted during TS 10-4 Phase I was planned for TS 10-4 Phase II, except that the fuel bundles used during Phase II were AI-CE-Memo 25 half-length bundles with stepped end plate design. Five of the

Figure 21000-9. TF 10-4 Fuel Bundle Vibration Amplitude vs Distance from Bottom of Bundle (Bundles Aligned)

Figure 21000-10. Vertical Motion of Magnet No. 17 in TF-10 Fuel String

UNC

7680-251004

Figure 21000-11. TS 10-4 SAP-to-SAP Flash Weld Failure as Viewed Through Borescope

half-length bundles were banded and five left unbanded. In addition, a thermal baffle assembly (see Pressure Tube Development Section) was installed in the upper pressure tube extension so that a temperature profile of the upper extension in the end seal area could be checked.

UNC 2-11-67

7680-251005

Figure 21000-12. TS 10-4 SAP-to-SAP Flash Weld Failure

Shortly after startup of Phase II testing, a failure of the lower SAP-to-SAP flash weld was experienced. At the time of the failure, TF-10 was operating under steady-state conditions at 550°F and 187-psig test section inlet pressure. Loop shutdown was automatic due to a low liquid level indication. A detailed boroscopic examination of the failure indicated that the weld had parted approximately two-thirds of the pressure tube circumference. Figure 21000-11 shows the area of the flash weld as seen through the boroscope beginning at the start of the crack and working clockwise around the tube ID to the end of the failure. On completion of the boroscopic examination, the pressure tube was removed from the caandria tube. Figure 21000-12 shows the tube OD in the area of the failed flash weld.

A summary of the TS 10-4 test operations up to the time of the flash weld failure is shown in Table 21000-2.

Probable cause of the failure was apparent poor fusion of the weld. A detailed metallographic evaluation of the failure is being performed, following which a complete failure analysis report will be made.

TABLE 21000-2
TS 10-4 OPERATION

Operation	Startup		Shutdown		Total Hours
	Date	Time	Date	Time	
1. TS 10-4 Phase I — No bundles installed in pressure tube	8/9/66	1435	8/11/66	1255	46-1/4
2. TS 10-4 Phase I — Bundles installed in aligned position, 2 upper shield plugs, 2 banded bundles, 3 unbanded bundles	9/9/66	1600	9/30/66	1556	504
3. TS 10-4 Phase II — AI-CE-25 half-length bundles installed in aligned position, 2 upper shield plugs, five banded bundles, five unbanded bundles	12/9/66	1430	12/10/66	0430	14

D. FIVE CHANNEL TEST LOOP (TF-20)

Shakedown of TF-20 was initiated; the remaining construction will continue into the next reporting period.

During this reporting period, the following work was in process or completed.

1) The electrical contractor completed the wiring of the loop, and shakedown of the control circuits was initiated.

2) The steam service piping contractor satisfactorily completed all of the steam piping to Cell No. 2.

3) The air system, received as excess equipment from EOGR, was installed. This system can supply up to 400 scfm of dry air at 100 psig for the operation of test facilities in Building 2.

4) The installation of part of the thermocouple wire and copper tubing, 150 pressure gauges, DP cells and transmitters was completed. The TF-20 Control Room was essentially completed, and shakedown is in progress. A data logger was received as excess equipment from the EOGR and is being installed. The equipment is capable of scanning 320 channels every 2.2 sec and can provide a readout on punched paper tape every 10 sec. A picture of part of the TF-20 Control Room is shown in Figure 21000-13.

5) The insulation contractor completed insulation of the primary loop and the makeup and drain system. The insulation of the purification system and steam tracing line is approximately 50% completed.

6) The ventilation contractor satisfactorily completed installation of the Cell No. 2 ventilation system, which is capable of 15 air changes per hr under normal operation and 30 air changes per hr under emergency conditions.

UNC

Figure 21000-13. TF-20 Control Room

7680-25-1006

7) The TF-20 purification system that was installed in Cell No. 2 was removed and shipped to the Whiteshell Nuclear Research Establishment (WNRE). The system consists of two Attapulugus Clay Columns, 6 filters, and 10 miscellaneous carbon steel valves. All replacement equipment was ordered and should be received and installed during the next reporting period.

E. COMPONENT TEST LOOP (TF-50)

Testing was performed on a rolled transition joint (sandwich-type Model 15) and snout seals. Summaries of the test results are presented in the Pressure Tube Transition Joint Development and Refueling Machine Development sections.

The test facility was operated on a 24-hr basis during the testing period with satisfactory performance of all components until 10/25/66. Irregular performance of a Chempump (750°F) at this time led to loop shutdown for inspection of both pumps. Results of the inspection are as follows.

1. 400°F Pump (2000-hr Operation)

Excessive bearing wear resulted in the rotor can contacting the stator liner. This appears to have resulted from dissolved and entrained gas not being vented from the bearing cavities. An investigation is in progress.

2. 750°F Pump (2000-hr Operation)

Excessive pump bearing, housing, and impeller wear caused by a loose bearing retainer screw was observed. The pump manufacturer is staking the bearing retainer screws in both pumps to avoid this problem in the future.

Pump repairs by the Chempump Manufacturing Company are expected to be completed by mid-January. Both pumps are still under warranty.

Facility modifications accomplished during the reporting period are as follows.

- 1) Preliminary piping modifications for new heaters (100-kw and 25-kw) were completed.
- 2) Ventilation modifications required to maintain ambient temperatures with the addition of higher heating capacity were completed.
- 3) Flow meters and a control valve were installed.
- 4) Components required for facility pressure cycling capabilities were installed.

F. PRESSURE TUBE IRRADIATION TESTS

The WR1 Pressure Tube Irradiation Program will be accomplished in three phases: (1) pre-irradiation qualification and evaluation, (2) irradiation testing, and (3) post-irradiation evaluation. A test plan draft was written covering the first two phases in detail. Because the various paths that phase 3 could follow depends on the course of phase 2, the writing of phase 3 was delayed until some in-reactor testing is achieved.

A design report for the WR1 test assembly is being prepared; this will include a detailed stress analysis of components.

The first pressure tube assembly (EXP-WR1-2001) will have rolled transition joints and is scheduled for insertion in position F-7 of WR1 in June 1967. The second assembly (EXP-WR1-2002) will have hot pressed joints and will be inserted in position E-10, 6 to 8 months later.

Fabrication of EXP-WR1-2001 is approximately 30% completed. The scheduled completion date is March, 1967. Pre-irradiation tests of the completed assembly at the Windsor facilities are scheduled to start at the end of March, 1967. Component qualification tests will be initiated in January, 1967.

Eight SAP 895 tubes were extruded to approximately 16-ft lengths with nominally 3.250 in. ID. Tubes of two-wall thicknesses, 0.115 and 0.150 in., representing 80 and 60% of minimum stress to cause rupture, were extruded. Ultrasonic inspection did not reveal any indications of significant flaws in either the billets or the extruded tubes. All of the thick-walled tubes were extruded at 800°F. Two of the thin-walled tubes were extruded at 800°F, one at 900°F and one at 1000°F. All tubes appeared to be free from significant die marks, gouges, and other surface defects; however, the tubes extruded at higher temperature exhibited slightly rougher surfaces, probably due to reduced lubricant properties; further evaluation is in progress. Pertinent dimensional data are included in Table 21000-3. Stress-rupture tests will be performed on specimens from the tubes. If the thin-wall tubes fail to meet the expected stress-to-rupture properties, then the thick-wall tubes will be used in the WR1 irradiation experiments.

TABLE 21000-3
SAP PRESSURE TUBE DIMENSIONS

Billet No.	Tube No.	Extrusion Temperature (°F)	Full Length Maximum Bow (In.)	Inside Diameter		Wall Thickness	
				Maximum (In.)	Minimum (In.)	Maximum (In.)	Minimum (In.)
507	P20	800	7/32	3.252	3.253	0.158	0.148
517	P21	800	3/16	3.254	3.251	0.157	0.142
524	P22	800	3/16	3.256	3.251	0.158	0.149
534	P23	800	1/16	3.254	3.251	0.158	0.148
507	P24	800	5/16	3.256	3.249	0.121	0.111
517	P25	800	3/16	3.259	3.246	0.120	0.111
524	P26	900	5/32	3.253	3.249	0.120	0.115
534	P27	1000	5/16	3.254	3.248	0.118	0.111

^a Vidiage measurements

Pressure tube wall thicknesses were originally selected based on present WR1 operating conditions, as outlined in the WR1 manual. Late in the quarter, it was learned that modification of these operating conditons will occur in September, 1967. At that time, the inlet temperature will be increased from

640 to 662 °F, and the inlet pressure to the F-7 site will be 284 psig, considering pressure drop in the inlet feeders. If it is assumed that a 10% overpressure condition is possible during continuous operation, the inlet pressure may reach 314 psig. At 662°F, the allowable stress, as determined by the 100,000-hr stress-to-rupture life, is 5360 psi. A tube wall of about 0.100 in. is thus required to withstand the inlet conditions in the WR1 assembly. Utilizing an 0.010-in. wear allowance gives a minimum required tube wall dimension of 0.110 in. This is smaller than the minimum of the extruded tubing, indicating that this tubing will be adequate for the assembly even during modified operation of the WR1.

Work was initiated at the Chattanooga Metallurgical Laboratory for stress-to-rupture testing of SAP tubing from the lot fabricated for WR1 use. Confirmation that the tubing has properties similar to those published for the 10% Al₂O₃ alloy will be obtained from four tube burst rupture tests.

Under short time conditions, the tube should withstand a maximum surge tank relief pressure of 365 psig at the header trip temperature of 725°F. Maximum hoop stress under these conditions will be 6100 psi. This stress is less than 2/3 yield strength (8000 psi) and 42% of ultimate strength; consequently, short-time over-pressurization should not be a threat to continued operation of the assembly.

A feasibility study was performed regarding the possible insertion of a pressure tube assembly in the U-3 loop of the NRU reactor at Chalk River. The study revealed that it is mechanically feasible to prepare this assembly but that certain aspects of the design will require development effort. These include (1) determining a method to produce a tapered section in the SAP tube at the lower end of the assembly for the transition joint in order to clear the calandria, and (2) performing an evaluation of possible vibration and wear problems associated with a 31-ft unsupported test section length, 26-1/2 ft of which is SAP tubing.

It is not feasible to have this assembly available for insertion in October, 1967 to coincide with the start of the U-307 experiment. A preliminary PERT network indicates an elapsed time of approximately fifteen months from start of design to date of insertion.

G. SAP IRRADIATIONS

Early in the report period all effort on the SAP irradiation capsules (NAA-84) was terminated in preparation for an anticipated transfer of test responsibility to PNWL. Long lead-time items, such as instrumentation system components, which had not been purchased were cancelled.

Late in the report period, direction was received to reactivate the work on the NAA-84 test series. The reinstituted test series is to consist of two capsules instead of the original four. This will require a reduction in the scope of the test series. As of the close of the report period, a statistical analysis of the impact of this reduction in scope on the test variables was under way. Purchase orders for long lead items were being reactivated.

H. LOOP COOLANT CHEMISTRY AND LEAK DETECTION

TF-50 was on-line during the first month of this quarter. A total of eleven samples was taken, the first nine during actual loop operation at 750°F, the last two since loop shutdown. Table 21000-4 summarizes the chemistry of these samples.

A flash point tester and autoignition test equipment were installed and standardized; flash point and autoignition temperature determinations are being performed as a routine service on coolant samples. Collection, measurement, and identification of the total gas contained in a coolant sample is being performed.

Three benzene bubblers from the TF-10 hydrocarbon analyzer sample-bypass lines were submitted to Chemistry for gas chromatographic analysis of their contents. No HB-40 was detected in the samples although a compound with an elution time between that of benzene and biphenyl was detected but not identified.

The hydrocarbon analyzer used for organic leak detection on TF-10 failed. The trouble was traced to a diaphragm rupture in the vacuum pressure pump. A replacement pump was obtained and the analyzer restored to working order.

A coke sample from the Goodrich seal and film scrapings from the CANDU mechanism were submitted to Chemistry for analysis. The data from the chemical analysis performed are given in Table 21000-5.

TABLE 21000-4
TF-50 COOLANT CHEMISTRY SAMPLE ANALYSIS

Sample No.	Date Isolated	MST (ABS/mg x 10 ⁵)	ASH (%)	HB (%)	Terphenyl Composition (%)			Biphenyl (%)	Flash Pt (°F)	Total Gas Volume (ml)
					Ortho	Meta	Para			
32	9/8/66	63.15	-	-	55.5	25.0	3.2	3.0	-	-
33A	9/12/66	61.83	107 x 10 ⁻⁴	-	53.9	26.8	3.0	2.6	-	150
33B	9/20/66	172.83	277.5 x 10 ⁻⁴	12.35	53.1	27.2	3.9	2.0	343	149.5
34	9/29/66	172.88	131 x 10 ⁻⁴	13.0	54.0	29.5	3.9	3.6	340	277
35	10/6/66	228.3	82 x 10 ⁻⁴	12.9	-	-	-	-	-	-
36	10/8/66	258.3	64 x 10 ⁻⁴	13.8	-	-	-	-	321	82
37	10/14/66	223.0	47 x 10 ⁻⁴	14.2	-	-	-	-	343	127
38A	10/20/66	-	2136 x 10 ^{-4(a)}	16.8	49.4	27.3	2.1	4.8	342	-
38B ^(b)	10/20/66	-	341.5 x 10 ⁻⁴	16.9	49.4	26.7	2.2	5.2	321	-
39 ^(c)	11/28/66	-	1060 x 10 ⁻⁴	11.2	54.6	28.2	2.0	4.1	355	-
40 ^(d)	11/28/66	-	1170 x 10 ⁻⁴	14.0	51.2	28.9	1.0	4.8	345	-

(a) The high ash concentration of TF-50-38A and successive samples is attributable to contamination resulting from installation of a new spool-piece.

(b) Sample taken directly from coolant circulation pump

(c) Sample taken from dump tank

(d) Sample taken from heater

TABLE 21000-5
B. F. GOODRICH AND CANDU COKE ANALYSIS

Date	Sample	Melting Pt (° C)	Fe (ppm)	Infrared Spectrum
10/25/66	Coke	65	4000	Identical to Spectrum of High Boiler from TF-50
10/25/66	Film Scrapings	55	-	-

Apparatus was purchased and assembled for TF-20 leak detection support. Hydrocarbon analyzer standardization was completed. Analyzer calibration was terminated due to system blockages caused by condensation of Santowax OM. A report covering the problems encountered during calibration operations is being prepared.

III. NEXT REPORT PERIOD ACTIVITIES

1. Shakedown of TF-20 with HB-40 will be completed, and the test sections TS 20-1A and TS 20-4 will be installed for the start of testing.
2. Fabrication, instrumentation, and installation of TS 20-1A and TS 20-4 will be completed.
3. The fabrication of the EXP-WR1-2001 irradiation test section will be completed. The EXP-WR1-2001 pressure tube assembly will be installed in TF-10 and preirradiation qualification tests will be initiated.
4. TS 20-5 fabrication will be initiated.
5. Phase II testing of TS 10-4A pressure tube assembly will be completed.
6. Reliability demonstration requirements for the pressure tube assembly will be finalized and reported in the form of a Technical Information Document.
7. The U-3 pressure tube design will be initiated.
8. Guide bearing tests for material selection will be initiated.
9. The TS 10-2 test report, a failure report on TS 10-4 pressure tube assembly, and an interim report on pressure tube internal wear, will be completed.
10. Installation of the required TF-50 primary coolant pumps will be completed, and the thermal cycling portion of the transition joint runoff test will be initiated.
11. Design of the NAA-84 test series will be completed based on the two-capsule approach. Fabrication of test assembly components and instrumentation and control systems will be initiated.

-

-

-

-

-

HWOCR QUARTERLY REPORT	
TITLE: Transition Joints	WBS NO: 21200
REPORTING PERIOD: October through December 1966	

I. PROJECT OBJECTIVES

The objective of this task is to develop a transition joint between the in-core and out-of-core sections of the HWOCR pressure tube assemblies. The joints must maintain strength and leak-tight integrity under all anticipated conditions of reactor operation. The material selected for the in-core pressure tube is SAP and the material to which it must be joined is 347 stainless steel. The transition joints being developed for this purpose are:

- a) SAP to 347 SS hot pressed joints
- b) SAP to 347 SS rolled joints.

The operating environment of the pressure tube includes internal pressure and flow of organic coolant at temperatures up to 765°F, thermal cycling, pressure cycling, vibration, and radiation. The pressure tube will also be subjected to piping reactions and other mechanical loads. The principal potential problem areas associated with transition joints are: (1) multiaxial thermal and mechanical stresses and (2) formation of brittle intermetallics at the joint interface by diffusion.

An intensive test and evaluation program for joints which appear most promising is being conducted, including mechanical and metallurgical evaluation. The transition joints will also be tested as parts of pressure tube assemblies in out-of-pile loops, WR1, and NRU.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION OF WORK TO DATE

A. SAP-STEEL HOT-PRESSED TRANSITION JOINTS

1. Testing and Evaluation of Joints

The modification of a test rig for thermal cycling transition joints in a molten salt bath medium was completed. Two joints (H-41 and H-42) were subsequently thermal-cycled 1000 times between 565 and 765°F. The peak heating and cooling rates were approximately 30°F/sec; however, the average rate was 4°F/sec over the 200°F range. After thermal cycling, the joints (now helium leak-tight) were tensile tested at 750°F, giving typical as-bonded strengths, as summarized in Table 21200-1.

TABLE 21200-1
SUMMARY OF TENSILE DATA FOR THERMAL-CYCLED
SAP-STEEL TRANSITION JOINTS

Joint No.	Thermal Cycling Data			Tensile Test Data		
	Cycling Media	Temperature Range (°F)	No. Cycles	Test Temperature (°F)	Load (lb)	UTS (psi)
H-41	Molten salt	565 to 765	1,000	750	26,650	10,800
H-42	Molten salt	565 to 765	1,000	750	27,350	11,100
H-40	Santowax R-argon	650 to 750	10,000	750	22,500	9,100
H-17*	Santowax R-argon	650 to 750	10,000	750	25,000	10,100
H-24*	Santowax R-argon	550 to 750	1,100	750	26,200	10,600

*Previously reported in AI-CE-54, "HWOCR Quarterly Technical Progress Report," April through June 1966

A transition joint (H-40) was thermal cycled 10,000 times between 650 and 750°F in a Santowax R-argon medium in the organic thermal cycling test rig. Heating and cooling rates were 2.7 and 0.9°F/sec, respectively. The joint was found to remain helium leak-tight. A tensile test at 750°F gave similar results, as shown in the table.

2. Test Section for TF-20

The fabrication of a test section (TS 20-4) for the TF-20 organic loop facility was completed. Hot-pressed transition joints were used. The hot-pressed transition joint (347 SS-SAP 895) was flash-welded to a SAP 895 pressure tube (4.000 in. ID x 0.105 in. wall). The flash weld parameters used were the same as used previously for test sections TS 10-4. The 253-in. -long TS 20-4 assembly is shown in Figure 21200-1.

Prior to flash-welding the transition joint to the TS 20-4 test section, a flash-welded comparison joint was made in 4.000 in. -ID x 0.100 in. - wall SAP 895 tubing for setup purposes. Tensile test specimens were machined from this comparison joint and tested at room temperature and at 750°F. The data from this test are presented in Table 21200-2. The data indicate a joint strength very near the strength of the parent tube.

Final checkout of TS 20-4 consisted of a hydrostatic proof test at 790 psig for 20 min, followed by a helium leak test, with satisfactory results.

3. WR1 Pressure Tube Subassembly

12-19-66 UNC 7680-40158

Figure 21200-1. TS 20-4
Pressure Tube Test
Section

The tooling for fabricating the hot-pressed transition joints for the WR1 subassembly was procured. A trial lot of ten flash-welded SAP 895

tubular joints was procured and is currently being evaluated mechanically and metallurgically. Fabrication of this subassembly is scheduled for completion during the next quarter.

4. Rod-Type Transition Joints

Fabrication of rod-type transition joints for both the WR1-954 and -953 irradiation experiments has been completed. Tensile test results are given in Table 21200-3 for control specimens having nickel-niobium diffusion barriers.

TABLE 21200-2

TENSILE TEST DATA FOR FLASH-WELDED SAP 895
4.000-IN. ID x 0.100-IN. WALL TUBING

Specimen* No.	Test Tempera- ture (° F)	Yield Strength 0.2% Offset (psi)	Ultimate Tensile Strength (psi)	Location of Failure
1	750	12,300	12,600	Through weld
2	750	11,800	12,500	Parent metal
3	750	11,900	13,000	Through weld
4	750	13,100	13,800	Through weld
5	RT	29,900	38,400	Through weld
6	RT	27,000	39,500	Through weld
7	RT	27,800	39,600	Through weld
8	RT	28,100	39,400	Through weld

*Tensile coupons 0.250 in. wide, 1.00 in. gage length

TABLE 21200-3

TENSILE TEST DATA FOR 0.350-IN. -DIAMETER ROD-TYPE TRANSITION
JOINT CONTROL SPECIMENS FOR WR1-953 AND -954

Specimen No.	Diffusion Barrier	Test Tempera- ture (° F)	Ultimate Tensile Strength (psi)
3CCC	Nickel-Niobium ↓	RT	33,000
2EEE		750	8,930
3GGG		750	7,910
3III		RT	33,200
1KKK		RT	32,400
4MMM		750	7,830
3PPP		RT	31,800
3QQQ		750	8,590

Comparable data on rod-type joints with nickel-tungsten diffusion barriers were given in the previous quarterly report. The appearance of the fractures in the tensile specimens, is similar to the fractures in tubular transition joints, in that bonded SAP material generally remains on the steel cone except for the extreme tip.

5. Pressure Tube Subassembly

It is recognized that the flash-welded joint, as presently used in pressure tube fabrication, is the weakest joint in the entire assembly. Figure 21200-2 illustrates the present concept and two alternatives. One alternative is to move the flash weld into a thickened end section (Figure 21200-2b). The other concept is to eliminate the flash weld by direct hot-pressing the stainless-steel transition joint component into the SAP pressure tube, as shown in Figure 21200-2c. Fabrication efforts are now directed at evaluating both these alternatives.

The alternate subassembly concepts of Figure 21200-2 require a SAP pressure tube with increased wall thickness at the ends. Methods for obtaining the desired wall thickening of the SAP tube generally fall into three categories:

- a) Hot-upsetting a thin-wall tube,
- b) Cold-forming a thin- or thick-wall tube,
- c) Metal removal from a thick-wall tube.

Cold-forming of the SAP tube by the use of explosives has failed to produce satisfactory results. Procurement action was initiated for obtaining hot-upset tubes by forging and cold-formed tubes by flow forming.

An initial evaluation of metal removal by chemical milling was completed, with very promising results. A 10-in. length of XAP 001 tubing was machined to cross-sectional dimensions of 4.000-in. ID x 0.187-in. wall. Masking was applied over a length of 2 in. at each end to retain the original wall thickness. The intent was to uniformly remove about 0.062-in. of material from the outside of the 0.187-in. -thick wall over the 6-in. length of unmasked SAP material. Measurements of the wall thickness along the chemically milled length of the sample revealed that 0.054 in. of material was removed from a distance 1/2 in. from the masking at each end. At a distance one inch from the masked ends and throughout the remainder of the chemically milled area, the metal removal

a. PRESENT CONCEPT (THIN WALL FLASH WELD)

b. ADVANCED CONCEPT (HEAVY WALL FLASH WELD)

c. ELIMINATION OF FLASH WELD (BY DIRECT HOT PRESSING)

2-10-67 UNC

7680-25982

Figure 21200-2. Present and Alternate Concepts of the
Pressure Tube Subassembly

was uniformly 0.065 in. The surface finishes of the machined and chemically milled portions of the tube were 50 to 55 and 70 to 75 micro-inches rms, respectively. The chemically milled specimen is shown in Figure 21200-3.

A distinct advantage of using a metal removal technique for direct hot-pressed subassemblies (Figure 21200-2c) is the ability to grip a heavy-wall tube (instead of a thin-wall tube) during hot-pressing of the transition joint. Gripping of the tube is considered a basic problem area in direct hot-pressing. Effort is being initiated on the design of tooling required for direct hot-pressing of the stainless-steel transition to the pressure tube.

Flash-welding trials are currently being conducted on heavy-wall SAP tubes in accordance with the conceptual requirements of Figure 21200-2b. This effort has necessitated locating a new supplier with a flash-welding machine of sufficient capacity to produce the heavier weld. The work is being done on a 800-kva Taylor-Winfield machine. Previous flash-welds on thin-wall SAP material were made on a 300-kva machine.

6. Nondestructive Test (NDT) Development

Nondestructive test development activities were directed towards (1) improving the method for the thermal deposition tungsten layer thickness measurements, and establishing the feasibility of a method for the electroplated nickel layer thickness measurement; and (2) establishing a method for evaluating the SAP-to-SAP flash-weld internal soundness.

Improvements of the speed and accuracy of tungsten thickness measurements for 0.0002-in. to 0.0006-in. layers has been accomplished. This was done by using an integral Sr^{90} source and detector probe assembly in conjunction with a constant position fixture. A 0.090-in.-diameter beta shield on the face of the probe provides an area of satisfactory size for thickness measurement, while permitting adequate count information to be accumulated in a reasonable time (approximately 1-1/2 minutes for each thickness measurement). Thickness standards have now been qualified by metallography at the 0.00025-in., 0.00045-in. and 0.00064-in. tungsten layer thickness for 347 SS, using the 60° "knife edge" and 4.000-in. -ID x 0.187-in. -wall design. Verification results on three joint samples measured nondestructively and then measured by sectioning and metallographic examination gave the results shown in Table 21200-4.

2-9-67 UNC

7680-40131

Figure 21200-3. Chemically Milled XAP 001 Pressure Tube Test Specimen

TABLE 21200-4
CORRELATION OF NONDESTRUCTIVE WITH DESTRUCTIVE
TUNGSTEN THICKNESS VALUES

347 SS Joint Section Sample Number	Location and Tungsten Thickness (mils)											
	0°		60°		120°		180°		240°		300°	
	NDT*	MET†	NDT	MET	NDT	MET	NDT	MET	NDT	MET	NDT	MET
B319120	0.25	0.22	-	-	-	-	-	-	-	-	-	-
B321007	0.43	0.45	0.38	0.38	0.43	0.44	0.66	0.59	1.00	0.75	0.63	0.65
B339825	0.49	0.51	0.52	0.51	-	-	-	-	-	-	0.43	0.44

*NDT - Nondestructive beta backscatter method

†MET - Metallographic section

For thicknesses up to 0.0065 in., no bias is evident between the two methods, and the accuracy of the beta backscatter method is estimated to be within $\pm 10\%$ of the value reported. Work is currently proceeding to qualify tungsten thickness standards at nominal values of 0.0010 in., 0.0015 in., and 0.0020 in.

A feasibility study demonstrated that the Hall effect method is suitable for measuring nickel layer thickness on 347 SS transition joint components. Three 347 SS joint sections were prepared for this study containing no nickel, 0.00007 in. to 0.00010 in. and 0.00014 in. to 0.00021 in. of nickel electroplated on the "knife edge." Figure 21200-4 shows the joint samples and test instrumentation employed, while Figure 21200-5 is a view of the Hall probe and one of the joint samples. Scans of the three samples using the Hall instrumentation are shown in Figure 21200-6. The sensitivity of this method to small differences in nickel thickness is evident from the scan spacing.

A contact ultrasonic technique was established for testing the SAP-SAP flash-weld joint. A plastic acoustic lens was made which refracts a shear wave at 89° , normal to the weld plane. A test standard was prepared from a 4.200-in. OD x 0.125-in. -wall XAP 001 tube. This standard contains inside and outside surface EDM notches, in the circumferential plane, measuring 0.030-in. wide and having depths of 0.005 in. and 0.010 inch. Testing must be accomplished from the outside surface. The current procedure detects the 0.005-in. outside

2-10-67 UNC

7680-25983

Figure 21200-4. Test Instrumentation, Fixturing, and Samples used to Demonstrate the Feasibility of Using Hall Effect to Measure Electroplated Nickel Layer Thickness on Pressure Tube Transition Joint Section

2-10-67 UNC

7680-25984

Figure 21200-5. Probe Assembly Containing Hall Device in Contact With Nickel-Plated "Knife Edge" on Stainless-Steel Section of Pressure Tube Transition Joint

Figure 21200-6. Hall Device Scans Around Periphery of 347 SS Joint Sections Illustrating Effectiveness of Method to Determine Electroplated Nickel Layer Thickness

surface notch and the 0.010-in. inside surface notch. This is done at near-maximum sensitivity, using a Sperry UM 715 Reflectoscope and a 2.25-MH test frequency with lithium sulphate transducers.

7. Diffusion Barrier Studies

Initial diffusion data were evaluated from couples containing Ni-W barrier coated on 347 SS and bonded to SAP 895. The activation energy calculated for these vapor-deposited tungsten-SAP 895 couples is greater than twice that reported previously* for tungsten rod SAP 895 couples. This result is attributed to the difference between the density of the vapor-deposited tungsten and that of the hot-pressed sintered tungsten rod. It is felt that surface diffusion occurring along particle boundaries in the tungsten rod is the reason for the low activation previously found. All current work is being conducted on couples fabricated with the vapor-deposited tungsten coating, i. e., the reference system.

*AI-CE-55, "HWOCR Quarterly Technical Progress Report," July through September 1966

Analysis of diffusion data for the vapor-deposited W-SAP 895 system has resulted in calculated values of $n = 0.75$ and $Q = 81,000$ cal/mol from the diffusion equation

$$x^n = k_o t e^{-Q/RT}$$

where

x = diffusion thickness (in.)

t = time (hr)

Q = activation energy (cal/mol)

These values are based on 11 data points in the 1000 to 1075° F temperature range. Additional diffusion couples are being isothermally heated to extend the temperature range of this study. On the basis of the 81,000 cal/mol Q value, a thermal treatment of only 105 hr at 950° F would be required to demonstrate equivalent 30-year diffusion at 765° F in SAP-steel transition joints with a tungsten diffusion barrier. Transition joints have been exposed for 500 and 1000 hours at 950° F. These tests represent severe overaging of the transition joints.

B. SAP-STEEL ROLLED JOINTS

1. Inner Sleeve Configuration

No further development or testing was done on this type of rolled joint during the last reporting period because of the inherent advantages of the sandwich-type joint to the reactor reference design. The information developed will be used for the calandria tube rolled-joint development program. This activity will no longer be reported.

2. Sandwich-Type Configuration

A summary of the SAP-stainless steel rolled-joint configurations fabricated during this report period is shown in Figure 21200-7. Design improvements made in each successive model are also noted. A summary of the test results of the joints tested during this report period is shown in Table 21200-5.

AI-CE-56
WBS 21200

Figure 21200-7. Summary of Rolled Joint Models

a) The double-ended Model 15 specimen fabricated in the last quarter was hot-tensile tested at 750°F, producing failure in the SAP tubing at a stress level of 15,000 psi (Figure 21200-8). Nominal tensile strength of SAP 895 tubing at 750°F is approximately 13,000 to 15,000 psi. Previous to hot-tensile testing, the specimen was soaked at 950°F for 500 hours, then thermal cycled 1000 times. The temperature range during the thermal cycling was from 550 to 800°F for the upper joint and from 400 to 800°F for the lower joint. There was no evidence of helium leakage through the joints before, during, or after any of the thermal tests. The helium leak testing was performed by using an instrument having a sensitivity of 10^{-9} cc/sec. A cross-section of the Model 15 joint is shown in Figure 21200-9. The stainless steel sections of the Model 15 joints were surface "oxidized" prior to rolling the joint, in a salt bath at 950°F for four hours to form a diffusion barrier. A metallographic examination of the two joints showed some evidence of intermetallic formation at the interface between the SAP tube and the inner SS sleeve. Figure 21200-10 shows a section of the inner sleeve/SAP interface where the oxide layer had apparently acted as an effective diffusion barrier. Figure 21200-11 shows a section of the joint near the closed end where direct SAP 895 to 304 stainless steel contact resulted in the formation of closely spaced intermetallics up to 0.002 in. thickness. The load applied during the tensile test had fragmented the intermetallics and had separated the SAP from the stainless steel. The structure of the intermetallics is duplex in appearance, suggestive that in addition to Fe_2Al_3 , there may be small amounts of other aluminum compounds.

TABLE 21200-5
TESTS OF ROLLED JOINTS

Specimen No.	Joint Size ID (in.)	Joint Type (Model)	Testing	Results
1	4.00	15	1) Helium leak test 2) Thermal soak 3) Thermal cycling 4) Hot-tensile test 5) Metallographic examination	1) Helium leak-tight when tested to 10^{-9} cc/sec instrument sensitivity 2) Thermal soaked for 500 hr at 950°F. Leak-tight after test 3) Withstood 1000 thermal cycles, top joint between 550 and 800°F and the bottom joint between 400 and 800°F. Maintained leak-tightness during and after test 4) Tensile tested at 750°F, broke in tube at 15,000 psi 5) Metallographic examination showed sporadic intermetallic buildups of up to 0.002 in. thickness
2	4.00	17	Same as Specimen No. 1	Same results except tube broke at 15,300 psi
3	4.00	15	1) Helium leak test 2) Thermal cycling in TF-50	1) Helium leak-tight to 10^{-9} cc/sec 2) Has withstood (a) 200 cycles at 30°F/sec between 400 and 735°F and (b) 200 cycles at 60°F/sec between 400 and 735°F, no observable leakage
4	4.00	15	1) Helium leak test 2) Will be installed in TS 20-1A for testing	1) No helium leakage when tested to 10^{-9} cc/sec
5	4.00	15	1) Helium leak test 2) Will be installed in TS 10-4A for testing	1) No helium leakage when tested to 10^{-9} cc/sec
6	4.00	19	1) Helium leak test 2) Thermal cycling 3) Hot-tensile test	1) Helium leak-tight to 10^{-9} cc/sec 2) Withstood 150 thermal cycles between 500°F and 800°F with no leakage 3) Tube broke midway between joints at a load of 17,000 lb at 750°F

2-10-67 UNC

7680-25987

Figure 21200-8. Double-Ended Model 15 and Model 17 Rolled Joints
Showing Type of Failure After Tensile Testing at 750°F.
(Failure occurred at a stress level of about 15,000 psi)

2-10-67 UNC

7680-25988

Figure 21200-9. Cross Section of a Model 15 Rolled Joint

2-10-67 UNC

500X

7680-25989

Figure 21200-10. Interface Between SAP Tube and Inner Stainless Steel Sleeve. (Oxide layer acted as effective diffusion barrier).

2-10-67 UNC

75X

7680-25990

Figure 21200-11. Intermetallic Formation at Interface Between SAP Tube and the Inner Stainless Steel Sleeve Near Closed End of Joints

The interface between the outer sleeve and the SAP tube exhibited a discontinuous oxide layer over the entire sleeve surface, with localized formation of intermetallics. It was not established whether the discontinuous nature of the oxide layer on the outer sleeve was due to the rolling operation or to the removal during the tensile test. However, it appears that the oxide layer may have been destroyed in regions of severe metal deformation from the rolling operation allowing metal-to-metal contact for intermetallic formation during the thermal treatments.

b) The Model 17 double-ended rolled joint specimen was tensile tested at 750°F, producing failure in the SAP tubing at a stress of 15,300 psi. The specimen was previously subjected to the same tests as the Model 15 specimen. No helium leakage was observed through the joints before, during, or after the thermal tests using an instrument sensitivity of 10^{-9} cc/sec. A cross-section of the joint is shown in Figure 21200-12. The stainless-steel sections of this specimen were soaked in a salt bath at 950°F for four hours to form an "oxidized" diffusion barrier. Metallographic examination showed a surface oxide layer as well as localized intermetallic formations similar in appearance to those in the Model 15 specimen. The inner sleeve surface in contact with the SAP tube showed essentially a continuous, dark grey "oxide" layer, about 0.0001 in. in thickness, and the complete absence of intermetallic formation. The contact surface of the outer sleeve showed discontinuous oxide, with the major portion of the surface devoid of the oxide layer. However, the presence of very shallow grain boundary oxidation indicates that the oxide layer may have been removed during the tensile test. Intermetallic formation up to 0.001 in. in thickness was limited to one location on the outer sleeve-SAP tube interface. Figure 21200-13 shows the nature

2-10-67 UNC

7680-25991

Figure 21200-12. Cross Section of Model 17 Rolled Joint

2-10-67 UNC

150X

7680-25992

Figure 21200-13. Characteristic Growth Patterns of the Intermetallics at the Interface Between the Outer SS Sleeve and the SAP Tube

of SAP-to-stainless steel interface as well as the characteristic growth pattern of the intermetallics. The shear action of the tensile test* had developed an irregular mode of failure within the SAP at a short distance from the interface. The SAP is adherent to the stainless steel in spite of a thin (less than 0.0001 in.) layer of intermetallics, Figure 21200-14. This interfacial layer is considered to be an intermetallic rather than an oxide on the basis of (1) light grey color similar to that of the more massive intermetallics, (2) the preferential growth into SAP, and (3) the highly adherent nature of the layer to both the SAP and stainless steel. An apparent reaction zone noted within the SAP of the Model 15 joints (Figure 21200-11) was not observed in the Model 17 joint.

c) The Model 15 tandem specimen installed in the TF-50 organic loop for thermal cycle evaluation has undergone the following thermal cycles:

- 1) 200 cycles at 30°F/sec between 400 to 735°F
- 2) 200 cycles at 60°F/sec between 400 to 735°F

Testing of this joint is continuing.

d) A pressure tube assembly was fabricated for the TS 20-1A test section, using two Model 15 transition joints. The joints were helium leak-tight to 10^{-9} cc/sec after assembly. The SAP 895 tubing used is approximately 20 ft long.

*At the open end of the joint

2-10-67 UNC

1000X

7680-25993

Figure 21200-14. Closeup of Intermetallic Shown
in Figure 21200-11

e) Due to a failure in the SAP-SAP flash weld of the TS 10-4 test section, a substitute tube designated TS10-4A is being fabricated using the Model 15 rolled joints. The joints will be rolled onto a 19-ft-long x 0.095-in. -wall SAP 895 tube. The tube assembly will be installed into TF-10 for completion of the tests planned originally for TS 10-4.

f) A tandem pair of Model 19 rolled joints were assembled using a 0.0015-in. tantalum foil diffusion barrier on one end and an "oxide" barrier on the other end. The joints were helium leak-tight as-rolled and after 150 thermal cycles between 500 and 800°F. Tensile testing at 750°F produced a failure in the SAP tubing midway between the joints. The stress on the SAP at failure was 14,5000 psi.* No apparent damage resulted in either joint. This demonstration of pullout strength and leak tightness gives considerable confidence that either the foil type of barrier or an "oxide" barrier can be used. Additional testing on the Model 20 joints is underway to confirm these early observations.

3. Diffusion Barrier Studies

A number of simple diffusion couples have been assembled as part of the rolled joint diffusion barrier investigation. The couples were made using discs of SAP and stainless steel bolted together. Several barrier materials were

*Joint Efficiency = 100%

used to separate the SAP and stainless steel. These included tantalum and tungsten foils, and various thicknesses of oxide films on both the SAP and stainless-steel surfaces. The couples are being held at 850, 950 and 1025°F for 100, 400 and 1600 hours.

Initial metallographic examination of couples held at three temperatures for 100 hours and 400 hours indicate that tantalum foil is compatible with both SAP and stainless steel. Concern exists about the tendency of tantalum to form hydrides in contact with organic coolants at 750°F. This problem has been examined briefly in a series of test exposures wherein 0.0015-in. tantalum foil was exposed to HB-40 at 750°F for times up to 70 hours with no evidence of a reaction. This study is continuing; however, it appears that hydriding of tantalum will not be a problem below 1200°F.

Based upon early results of the diffusion barrier study, both tantalum and a continuous oxide film on the stainless steel were found effective in eliminating SAP-stainless steel reactions. The major limitation on the effectiveness of the oxide diffusion barrier was the occurrence of breaks in the oxide film at sharp corners in the interface. To eliminate such corners, the grooves of several Model 15 joints were modified to a smooth sinusoidal contour having approximately an 0.008-in. amplitude. This modification is referred to as the Model 19 joint.

III. NEXT REPORT PERIOD ACTIVITIES

A. SAP-STEEL HOT-PRESSED JOINTS

Activities planned for the next report period include the following:

- 1) Continuation of diffusion barrier studies,
- 2) Continuation of NDT development for control of Ni-W coating thicknesses and for determining integrity of flash-welded SAP and hot-pressed transition joints,
- 3) Completion of fabrication of the WR1 pressure tube subassembly,
- 4) Continuing evaluation of flash-welded SAP joints,
- 5) Continuing development of SAP pressure tubes with increased wall thickness at the ends,

6) Initiation of development of direct hot-press bonding of transition joints in pressure tubes (to eliminate flash welding),

7) Fabrication of test articles for the TF-50 loop for comparative tests with rolled transition joint test articles.

B. SAP-STEEL ROLLED JOINTS

1) Continuation of testing and evaluation of diffusion barrier,

2) Continuation of the fabrication and tests of the Model 20 rolled joint,

3) Fabrication of test specimens for the comparative tests with the hot-pressed joint test specimens.

1

2

3

4

5

6

HWO CR QUARTERLY REPORT

TITLE: Pressure Tube Fabrication Development*

WBS NO: 21800

REPORTING PERIOD: October through December 1966

I. OBJECTIVES

The objective of the pressure tube fabrication development task is to qualify supplier(s) for the fabrication of SAP pressure tubes. To achieve this objective, it will be necessary to develop a SAP pressure tube fabrication procedure; develop a SAP pressure tube product specification; establish nondestructive testing methods; and evaluate the results to assure SAP pressure tubes meet the requirements for HWO CR plants.

The objective will be met by furnishing SAP billets to the extrusion vendors. The qualification of supplier(s) for producing SAP billets is not a part of this task; however, the pressure tube fabrication development work will be coordinated with the SAP billet source development work.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

Work was started towards the development of the fabrication techniques required to produce SAP pressure tubes for HWO CR. Billet material which can be converted to pressure tubes by extrusion was received from Alcoa and Montecatini. This work will be coordinated with the SAP billet procurement activity to eventually permit an integrated process for the fabrication of pressure tubes, starting from aluminum powders. Billets presently on hand are being used in order to permit extrusion development work in parallel with the development of SAP billet sources. An important by-product of this extrusion work is the manufacture of tubes for the in-pile and out-of-pile loop testing being performed under pressure tube development tasks such as 21100 and 21600.

*A new task, not included in previous quarterly reports.

Some SAP tubes were extruded by Nuclear Metals Incorporated, using the surplus billets received from AECL. These tubes were nominally 4-in. ID, 0.090 in. wall thickness and 30 ft in length. The initial extrusion trials resulted in stalled extrusion. Minor modifications to die design, extrusion temperature, lubrication procedure, and ram speed alleviated the problem and eleven full length tubes were extruded in this size range (lots 1 and 2). Since several parameters were altered simultaneously, the critical parameters could not be identified. Subsequent inspection of these tubes revealed several surface defects. Ultrasonic inspection of the tubes from lot 1 was performed at CBL Laboratories in Cleveland, Ohio. The results of the ultrasonic inspection were considered questionable because of a lack of correlation with visually observed defects. The inspection data are shown in Table 21800-1. The results of mechanical property tests for this lot of tubing are shown in Tables 21800-2 and 21800-3. The axial and hoop properties in general, insofar as yield and tensile strength are concerned, are in agreement with the values published in the SAP handbook.* The hoop ductility values are lower than the handbook values and the spread in the ductility data is large for both axial and hoop tests.

A second lot of four 4-in. diameter SAP tubes was extruded using the same extrusion parameters that were developed for the first lot. Six extrusion attempts were made, one resulted in a split mandrel and one in a stalled extrusion. Two of the four tubes extruded, tube numbers P-14 and P-16, were XAP 001, and two were SAP 895, tube numbers P-17 and P-18. The dimensional measurements of this tubing are shown in Table 21800-4.

Tube straightening was performed at Chase Brass in Waterbury, Connecticut using support shoes and a hydraulic jack to apply lateral loads for bending. It is obvious that, although any bow in the tubes was virtually eliminated, this method did result in considerable ovality. A 3.950-in. diameter plug gage would not pass through any of the four tubes (minimum ID measured 3.994 in.). Three of the tubes were ultrasonically inspected at TAC in Plainfield, New Jersey. The results of the inspections are shown in Table 21800-5. Some of the ultrasonic indications of defects show the defect as having a transverse orientation in the tube wall. Visual and mechanical testing of this lot is currently underway.

*"SAP Handbook," AI-CE-Memo-24, March 23, 1966

TABLE 21800-1
SAP PRESSURE TUBE INSPECTION SUMMARY
(Lot Number 1)

Selection Number	Tube Number	SAP Billet	Number of Defects in SAP Billets	Visual Defects in SAP Tube	Ultrasonic Defects Noted in SAP Tube	Recommended Disposition	Wall Thickness Range (in.)
1	P-12	Center of Wo2A	1 flaw 3/4 in. from surface	Kinks located 3 ft from front end	4 defects <0.006 in. located over first 5 ft of tube	Use for 1st rolled joint. Remove 1st 3 ft then roll first joint and remove excess from opposite end	0.102 to 0.112
2	P-11	Rear of Wo3A	None	External blisters first 7 ft, slight kink at 1 ft mark, score marks at 4 ft mark	2 defects <0.006 in. located at 1 foot from each end	Remove 7 ft from front end and send the remaining 22 ft to AI for hot pressed joint	0.103 to 0.110
3	P-8	Front of Wo2A	1 flaw 3/4 in. from surface	External blisters located from 7 to 17 ft mark	No defects noted	Remove 9 ft from front end and send the remaining 20 ft to AI for hot pressed joint	0.098 to 0.104
4	P-10	Front of Wo3A	None	0.006 in. pit at 9 ft mark, 0.009 in. gouge at 16 ft mark, 0.003 gouge at 21 ft mark	No defects noted	Use as backup for roll joint. Cut only if necessary then remove front 9 ft, if possible	0.100 to 0.105
5	P-6	Rear of Yo6	Not tested	Inclusions at 19 ft mark	7 defects over first 8 ft of tube length. 2 defects <0.014 in., 1 defect <0.011 in. remainder <0.006 in.	Use as second backup to roll joint or cut into tube burst samples if not required for rolled joints	0.097 to 0.106
6	P-7	Center of Wo3A	None	Hole 0.033 in. at 9-1/2 ft marks	4 defects over first 4 ft of tube length. 2 defects <0.011 in., 1 defect <0.009 in., 1 defect <0.006 in.	Cut up for tube burst tests	0.105 to 0.113
7	P-9	Front of Wo6A	None	Pits, laps, score mark, blisters and inclusions located over most of tube length	9 defects over most of tube length, 2 defects <0.023 in., 2 defects <0.017 in., 4 defects <0.011 in., 1 defect <0.006 in.	Selective cutting could produce a few tube burst tests. Metallurgical and mechanical tests should be performed on defected regions	0.099 to 0.105

TABLE 21800-2

TENSILE DATA ON SAP 895 PRESSURE TUBES (Lot Number 1)
(All Tests at Room Temperature on Flat Strips Cut
Longitudinal to Tube Axis)

Sample from Tube	Tensile Strength (psi)	Yield Strength (psi)	Elongation (%)	Crosshead Speed (in./min)
P-4	41,500	28,600	14	0.05
P-4	41,100	28,600	13	0.05
P-4	41,600	28,000	8	0.05
P-4	42,100	28,900	11	0.05
P-4	42,000	28,500	13	0.05
P-4	42,200	29,400	11	0.05
P-6	42,300	28,300	11	0.05
P-6	42,400	29,800	13	0.05
P-6	41,900	29,000	10	0.05
P-7	41,100	27,800	12	0.05
P-7	41,300	28,000	12	0.05
P-7	41,300	28,000	12	0.05
P-8	40,500	27,000	12	0.05
P-8	40,400	27,000	9	0.05
P-8	40,800	27,900	12	0.05
P-8	41,500	28,700	14	0.05
P-8	41,715	29,000	13	0.05
P-8	41,300	29,200	11	0.05
P-8	40,300	28,600	14	0.025
P-8	41,600	29,000	13	0.1
P-8	41,100	29,500	14	0.025
P-9	42,200	29,600	13	0.025
P-9	41,800	29,400	11	0.025
P-9	42,000	28,900	12	0.1
P-9	42,600	29,600	8	0.05
P-9	42,100	28,100	8	0.05
P-9	41,600	28,900	12	0.05
P-9	41,700	28,900	12	0.05
P-9	41,700	28,800	12	0.05
P-10	41,900	28,300	12	0.05
P-10	41,600	28,200	12	0.05
P-10	41,700	28,000	12	0.05
P-10	42,600	29,400	11	0.05
P-10	42,700	28,800	7	0.05
P-10	42,300	28,500	8	0.05
P-10	42,600	29,000	8	0.05
P-11	42,100	28,900	15	0.05
P-11	41,900	28,500	14	0.05
P-11	42,300	29,800	14	0.05
P-11	42,000	28,900	12	0.1
P-12	41,900	29,600	13	0.1
P-12	41,300	29,200	7	0.025
P-12	41,600	28,900	14	0.1

TABLE 21800-3
HOOP DATA ON SAP 895 PRESSURE TUBES USING RING SPECIMENS
(Lot Number 1)

Sample from Tube	Group	Tensile Strength (psi)	Elongation (%)	Load Rate at Room Temper- ature in Air (in./min)
P-4	B	43,400	4	0.05
P-4	B	42,200	4	0.05
P-4	B	42,800	4	0.05
P-5	A	39,700	8	0.05
P-5	A	39,700	8	0.05
P-5	A	39,200	7	0.05
P-5	A	38,650	7	0.05
P-5	A	39,150	6.5	0.05
P-5	A	39,850	7	0.05
P-5	A	39,650	7	0.05
P-8	A	39,650	8	0.05
P-8	A	39,100	8	0.05
P-8	A	40,000	10	0.05
P-8	A	39,650	10	0.05
P-8	A	39,000	8	0.05
P-10	A	39,500	10	0.05
P-11	A	39,350	8	0.05
P-11	A	38,150	7	0.05
P-11	A	39,800	9	0.05
P-11	C	40,850	8	0.05
P-11	C	40,300	6	0.05

Group A - as received tube

Group B - hydraulically stretched 5% initially

Group C - explosively stretched 5% initially

TABLE 21800-4
SAP TUBING DIMENSIONAL MEASUREMENTS
(Lot Number 2)

Tube Number	Inside Diameter (in.)		Wall (in.)		Outside Diameter Before Straightening (in.)		Outside Diameter After Straightening (in.)		Maximum Bow (in./ft)
	Maximum	Minimum	Maximum	Minimum	Maximum	Minimum	Maximum	Minimum	
P-14	4.025	3.994	0.093	0.090	4.200	4.182	4.210	4.162	0.125/16
P-16	4.002	4.014	0.096	0.093	4.201	4.190	4.218	4.182	0.125/16
P-17	4.020	3.997	0.098	0.091	4.202	4.187	4.229	4.151	0.140/32
P-18	4.034	3.996	0.096	0.090	4.203	4.184	4.249	4.141	0.090/32

TABLE 21800-5
ULTRASONIC INSPECTION OF SAP
PRESSURE TUBES
(Lot Number 2)

Tube Number	Number of Defect Indications	
	Transverse	Longitudinal
P-14	10	7
P-17	38	9
P-18	21	12

The third lot of tubing extruded consisted of four tubes 3.250-in. ID by 0.155-in. wall and four tubes 3.250-in. ID by 0.115-in. wall. The dimensional inspection data for this tubing are shown in Table 21800-6, and the extrusion parameter data are shown in Table 21800-7. Visual inspection of this tubing shows some unusual ring-shaped shallow depressions in many areas on the OD surfaces of the tubes. The inner tube surfaces have bands of shallow depressions resembling chatter marks. Ultrasonic inspection of this tubing showed no defects. Further examination of this lot is underway.

AI-CE-56
WBS 21800

TABLE 21800-6

SAP PRESSURE TUBING DIMENSIONAL DATA FOR WR1
IRRADIATION EXPERIMENT
(Lot No. 3)

Tube No	Inside Diameter (in)				Wall Thickness (in)				Outside Diameter (in) Entire Length (2-ft intervals)	
	Front		Rear		Front		Rear		Maximum	Minimum
	Maximum	Minimum	Maximum	Minimum	Maximum	Minimum	Maximum	Minimum		
P-20	3 256	3 252	3 255	3 253	0 158	0 154	0 158	0 149	3 565	3 557
P-21	3 254	3 253	3 254	3 251	0 157	0 153	0 164	0 145	3 564	3 555
P-22	3 256	3 253	3 254	3 251	0 156	0 154	0 158	0 150	3 562	3 558
P-23	3 254	3 251	3 253	3 251	0 156	0 153	0 158	0 149	3 561	3 555
P-24	3 256	3 253	3 252	3 249	0 118	0 117	0 121	0 114	3 489	3 483
P-25	3 254	3 251	3 259	3 246	0 120	0 117	0 119	0 113	3 491	3 478
P-26	3 253	3 249	3 251	3 250	0 120	0 117	0 120	0 117	3 487	3 481
P-27	3 254	3 248	3 253	3 248	0 118	0 114	0 118	0 113	3 484	3 475

TABLE 21800-7

SAP PRESSURE TUBE EXTRUSION PARAMETERS
(Lot No. 3)

Tube No	Temperature (°F)	Running Force (tons)	Speed (ipm)	Extrusion Constant (tsi)
P-20	800	1200	100	1 33
P-21	800	1350	100	1 49
P-22	800	1075	100	1 19
P-23	800	1150	100	1 27
P-24	800	1250	100	1 27
P-25	800	1350 (stall)	100	1 37
P-26	900	1000	100	1 02
P-27	1000	900	100	0 92

The tube burst stress rupture work was resumed after the move of the laboratory at Chattanooga. Specimens from selected pressure tubes are being tested at temperatures between 700 and 900°F by internal gas pressure. Those specimens, tested at hoop-to-axial stress ratios of 2:1, are closed by mechanical fittings and held at temperature and stress until rupture. Those specimens tested at 1:1 biaxial stress ratio have an axial load applied by external fixturing. Table 21800-8 shows the data generated during the report period.

TABLE 21800-8
STRESS RUPTURE TESTS OF SAP 895 PRESSURE TUBING

Specimen Number	Tube Number	Hoop Stress (psi)	Hoop: Axial Ratio	Temperature (°F)	Time (hr)
1-3	P-8	8200	2:1	900	1500*
3-5	P-9	6650	2:1	800	200*
1-5	P-9	6650	1:1.026§	800	12†
1-5a	P-9	6200	1:1	800	To be started

*Still in test

†Ruptured transverse to tube axis

§Axial load intended to produce 1:1 biaxial ratio; error in loading made this value 1:1.026

Specimen 1-5, which ruptured in the transverse direction, has caused some concern and specimen 1-5a has been added to check the data from 1-5. The specimens are cut from the same tube and 1-5a will be loaded at 1:1 biaxial stress ratio. The data point for the 1-5 falls below the predicted value, but above the minimum design value for the tubing.

III. NEXT REPORT PERIOD ACTIVITIES

The tasks for the next period include the objectives listed below.

- 1) Completion of the evaluation of the three lots of tubing extruded by NMI.
- 2) Continuation of the creep rupture testing at Chatanooga.
- 3) Completion of the evaluation of proposals from suppliers involved in the pressure tube development program and a negotiation of contract with one supplier.
- 4) Continuation of the development work associated with the nondestructive testing of pressure tubes.

HWO CR QUARTERLY REPORT

TITLE: Reactor Control Systems

WBS NO: 22000

REPORTING PERIOD: October through December 1966

I. PROJECT OBJECTIVES

The objective of this task is the development of the reactivity control systems necessary for the safe operation of the reactor. In general, control systems must be provided to compensate for changes in reactivity associated with (a) fuel temperature change, (b) coolant temperature changes, (c) buildup and decay of xenon and samarium, (d) moderator temperature changes, (e) void effects in the coolant and the moderator, and (f) fuel burnup. In addition, an adequate shutdown margin must be provided.

The HWO CR Program goals of high fuel utilization and economic power costs require the design of the reactivity control devices to minimize:

- a) The excess reactivity needed for control purposes.
- b) The parasitic material remaining in the core at full power.
- c) In-core power peaking.
- d) Initial capital costs and replacement costs.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. GENERAL

The objectives listed below were completed during this quarter.

- 1) Layouts were completed on two designs of the double fixed rolling bearing concept.
- 2) A 300-lb double coil electromagnetic latch was designed and fabricated.
- 3) The test facility was modified to incorporate an automatic cycling system, a manifold and external pump to simulate the moderator velocity, 2 in. piping, and an air-actuated control valve.

- 4) The evaluation of the 5 and 10% borated foamalum samples was completed.
- 5) A bearing test rig was designed and fabricated to evaluate various bearing material combinations.
- 6) Evaluation of components with the shutdown control test device was continued.
- 7) Reliability analyses were made for shutdown, power and xenon control.
- 8) Most of the components for the power and xenon control rod drive test device were received and assembly of the test device was initiated.
- 9) A preliminary structural support analysis of the power, xenon, and shutdown control system was prepared.
- 10) An investigation of an alternate method of obtaining reactivity for xenon override was initiated.

B. SHUTDOWN CONTROL SYSTEM

1. Development Progress

During the quarter, layouts of two designs of the double-fixed rolling bearing shutdown control concept were completed. In both designs the poison element is supported on 2 sets of spring-loaded rolling bearings spaced 4-ft apart axially. Each set of bearings consists of 4 rollers oriented 90° apart with one stationary member and 3 spring-loaded members. A floating piston is attached to the end of the rod to act as a seal with the cylinder and to support the rod when fully withdrawn. A hollow poison element and a magnetic latch were incorporated in the designs. One design employs a latch configuration in which the holding magnet is located at the rear of the cylinder and a long extension, carrying the holding plate, is affixed to the piston; the other design employs a latch configuration in which the magnet is affixed to a tube which extends inside the compressed scram spring and makes contact with the holding plate attached directly to the piston.

Both these concepts were compared with the reference design using a sliding piston and rolling nose bearings. The main advantage of the double-fixed bearing is the minimization of the piston bearing load which minimizes the wear between piston and cylinder. However, this advantage is offset by the disadvantages of the double-fixed bearing concept which are that the concept (a) is more complex, requiring an additional set of rolling bearings and springs and an additional support when withdrawn; (b) requires design of rolling bearings to

function under both an up and down reaction load; (c) requires increased fabrication cost; and (d) requires increased control rod deflections.

Based on this evaluation, no further effort was undertaken on this concept, and a revised design of the reference concept, based on results of the test program, was initiated.

The evaluation of the 3 additional samples of 5 and 10% borated foamalum has been completed and a final report is forthcoming. The samples were subjected to radiographic inspection, density measurements, and B_4C content analysis.

A study was also initiated on the use of cadmium as the neutron absorber for the shutdown control rods. Cadmium may be used if a hollow poison element is selected as the reference design.

2. Test Program

A 300-lb double-coil latch magnet was designed and fabricated for the test actuator device. Curves of holding force vs current were generated with the aid of tensile test equipment and are shown on Figures 22000-1, -2, and -3. Evaluation of this prototype magnet indicates that it is capable of producing enough force to latch the rod when one coil is energized and a 0.015-in. gap exists between the holding plate and the magnet assembly.

Figure 22000-1. Load vs Current for Double Coil Magnet, Both Coils Energized

2-7-67 UNC

7680-25961

Figure 22000-2. Load vs Current for Double Coil Holding Magnet, Rear Coil Energized

2-7-67 UNC

7680-25962

Figure 22000-3. Load vs Current for Double Coil Holding Magnet, Front Coil Energized

A 125-gpm circulating pump discharging into a perforated 14-ft long manifold was added to the test facility to study the effects of turbulence on the withdrawal and scram of the rod. In addition, a 2-in. air-actuated ball valve was installed in the fluid transfer system and all plumbing was changed to use 2-in. piping. A timing switch and an automatic cycling system were installed in the test facility in anticipation of extensive wear tests.

During the quarter the following components were evaluated in the test device.

a) Hollow poison element — The wax was removed from the simulated control rod tube, the nose piece was perforated, and a 1-in. diam hole was drilled through the piston to provide a direct path for the D_2O from the moderator tank to the head end of the cylinder. These modifications eliminated the hesitation in the stroke as previously experienced with the test device.

b) Self lubricating bearings — One set of nose bearings fabricated from a 75% Ni - 25% WSe_2 material was purchased. The bearings were removed from the test device after experiencing 75 scrams and a 21-day soak in demineralized water. The exterior surfaces of all pieces showed evidence of serious corrosion and the bottom nose bearing had fractured into 3 segments. The material is extremely brittle and could not tolerate the shock loads imposed by the rod.

c) Haynes 25 - Stellite 19 nose bearings — Bearings fabricated from the Haynes 25 - Stellite 19 reference materials were installed in the test device. The bearings were examined after experiencing 25 scrams and showed indications of galling. Calculations indicated that the bearing load had been increased from the 120-psi design condition to 215 psi when the non-floating hollow poison element was incorporated in the test device. The load condition for the hollow poison element has been decreased in the reference actuator design by increasing the support bearing distance from 2 to 4 ft.

d) Piston — A sliding piston on which a carbon graphite bearing was installed has been fabricated and is under test.

A bearing test rig was fabricated to evaluate the compatibility of numerous bearing materials in the HWOCR environment. The test rig consists of a horizontal motor-driven shaft rotating in a tank of demineralized water. The various bearing and pin combinations were mounted on cantilever arms and allowed to contact the rotating shaft. Various weights were hung from the ends of the arms to simulate the different load conditions. To date the tasks listed below have been completed.

- a) Stellite 19 roller running against Stellite 6 pin. Completed 1×10^6 cycles at 90 rpm under a 60-lb load (225 psi).
- b) Stellite 19 roller running against Waukesha 88 pin. Completed 1×10^6 cycles at 90 rpm under a 60-lb load (225 psi).
- c) Stellite 19 roller running against a Haynes 25 pin. Completed 1×10^6 cycles at 90 rpm under a 60-lb load (225 psi).

Components were purchased to provide a 200°F clean water environment for the test rig.

3. Shutdown Control System Reliability Analysis

A reliability analysis of the shutdown control system was prepared. The reliability for the shutdown mode of operation was 0.99162. This value was based on a useful life of 30 yr for the shutdown control actuator with 3 permissible failures (59 rods out of 62 rods satisfactorily operate upon initiating a scram). Based on the results obtained it is planned that 6 prototype mechanical actuators be tested at a 50% confidence level. Such a test will require that each actuator be subjected to 10 cyclic lifetimes (2680 cycles) with zero failures. If the test program were set up to achieve a 90% confidence level, using 6 actuators, each actuator would be required to operate with zero failures 30 cyclic lifetimes. Such a test would require the actuator to function well into the wear-out phase which would invalidate the data obtained to calculate the reliability of the device. On the basis of economics, the testing of 6 actuators at a 50% confidence level appears to be the most prudent approach.

C. XENON OSCILLATION AND POWER LEVEL CONTROL

1. Development Progress

a. Control Rod Actuator Design

The horizontal configuration of xenon and power level control rod results in deflections of the long components such as the rack and the control rod. Several anti-friction rolling supports were provided for the rack in the vertical as well as in the horizontal plane to limit the rack deflection to an acceptable amount, decrease the actuating force and component wear, and increase the life of the rack and pinion drive.

Because of the guide tube deflection, the control rod element is designed as a segmented assembly of absorber elements (12-in. long, 1-1/4 in. OD) coupled by swivel joints. Two types of segmented absorber elements are being studied, a non-floating 0.200-in. thick wall cylinder and a near-floating 0.020-in. thick wall stainless steel tube internally clad with a 0.020-in. thick wall cobalt tube.

The general design of the horizontal power control rod actuator package is shown schematically in Figure 22000-4.

b. Vertical Support Column

A preliminary analysis of the structural support for the power, xenon, and shutdown control systems was made to size 9 in-core vertical support columns. These columns, which provide in-core end support for the power and xenon control rod horizontal guide structure, were sized so that they could be installed between pressure tubes with a minimum lattice pitch of 10 in.

The analysis considered the structural deflections of the reactor vessel and upper and lower shield tanks under normal steady-state operating conditions. Structural deflections that occur in the reactor vessel result from static loadings and thermal gradients. Thermal gradients across the reactor vessel are caused by the difference between inlet and outlet temperatures of the D₂O which are 140 and 200°F respectively.

Figure 22000-5, shows the basic conceptual design of a support column. The in-core portion of the column is fabricated from a 2-5/8-in. OD by 0.100-in. wall zircaloy-2 tube. The columns are supported by the upper shield tank and require a 5-1/4 in. clearance hole through this region. Each column contains its own shield plug and is instrumentated with 5 self-powered neutron detectors; the detectors are used to provide in-core monitoring of neutron flux. The lower end of each column obtains lateral stability by a sleeved joint (receptacle). The joint compensates for axial movement between support columns and reactor vessel resulting from relative differential growth due to thermal expansion. Such a joint is essential to prevent the support columns from carrying a portion of the axial loads borne by reactor vessel tubes.

Alignment features have been incorporated into the design of the support column to simplify the installation of the power and xenon control rods. The

Figure 22000-4. Horizontal Power Control Rod Schematic
(Showing One Section of the Tri-Foil Arrangement)

2-7-67 UNC

7680-25964

Figure 22000-5. Vertical Support and Instrumentation Column, Power and Xenon Control System

columns can be adjusted vertically 2-1/8 in. and rotationally oriented within $\pm 10^\circ$. Each receptacle supporting the end of a control rod assembly (tri-foil) is provided with a saddle and guide fins. This design configuration permits blind installation of control system assemblies. If necessary, this technique can also be used to replace entire control rod units during a scheduled maintenance period when the reactor is in a shutdown condition.

The analysis shows that during steady-state operating conditions the support columns nearest the circumferential shell of the reactor vessel will, in the worst case, deflect radially 0.195 in. This point of maximum deflection occurs approximately 140 in. below the lower surface of the upper shield tank.

c. Alternate Reactivity Control Method

The use of vertical columns containing D_2O , H_2O , or void (helium) is being investigated as a power control system for xenon override. This concept consists of dispersing control columns within the central 50% (by volume) core region of the reactor. The control columns would be located on a 20-in. pitch of a reactor having a pressure tube pitch of 10 in. Reactivity is varied by completely filling or voiding the tubes; fine control of the reactor is still accomplished by horizontal power control rods.

A preliminary investigation of the reactivity worth of this concept for control columns dispersed throughout the reactor having an infinite lattice cell was made. The pressure tube and fuel cluster geometry used in the analysis was similar to that of a previous analysis* (37 rods, 6 in.² smeared UC fuel area, organic volume fraction = 0.39, and fuel enrichment = 1.39%).

The control columns would be fabricated from zircaloy-2 tubing, 3-1/2-in. OD and 0.035-in. wall thickness. The columns are fabricated with integral formed circumferential stiffeners (convolutions having depth of approximately 3/8 in.). The pitch between convolutions is 10 in. over the length of column. This design provides sufficient stiffness to allow the columns to withstand an external working pressure of 42 psig (accident condition) with a 2.6 factor of safety. By using this method of increasing the column stiffness, in-core parasitics can be minimized as opposed to increasing the wall thickness of the column.

*op. cit AI-CE-Memo-25

The results of the preliminary physics analysis show that the reactivity gain is 0.3% by filling voided columns with D_2O ; filling the voided columns with H_2O results in a reactivity loss of 3.5%. Based on these preliminary calculations, it appears feasible to obtain a reactivity control worth of about 1% using H_2O -filled columns in the central 50% volume of the core. Further studies will be made to investigate methods of filling and voiding control columns as well as to consider safety and economics factors.

d. Analysis of Xenon Induced Spatial Power Oscillations

Documentation of the analysis of xenon-induced spatial power oscillations with 1- and 2-dimensional analytical techniques was initiated during the latter portion of the period. The 1-dimensional approach to the xenon oscillation problem, e.g. SOS-1 code,^{*} gives some insight to the functional dependence of the threshold of instability on such parameters as the magnitude of the assumed physical dimension, the migration area, the shape of the power distribution along this dimension, and the power coefficient. In addition it also provides some measure of the oscillation period as a function of flux level. The 2-dimensional approach, as embodied in the XENOSC code,[†] has been employed to define the variation in spatial power distribution with time following a localized perturbation in the xenon density. The latter calculations serve to define the spatial power in a plane transverse to the fuel channel for the case of uncontrolled oscillations; i. e. no control measures are introduced to restrict the build-up of power oscillations.

Xenon-induced oscillations are viewed primarily as an operational problem since there is evidence to indicate that they can be controlled. Therefore, one of the primary objectives of the analytical effort is to provide data for a selection of an optimized control system. The factors involved are the increased costs associated with increasing the number of control zones and the increased margin required for local power density variations as the number of control zones is

^{*}The SOS-1 code, received from D. Randall (SRL) by personal communication, is based on the articles of D. Randall and D. S. St. John, "Xenon Spatial Oscillation," Nucleonics, March 1958, pp 82-86 and "Xenon Spatial Oscillation," Nuclear Science and Engineering, Vol 14 pp 204-206

[†]R. M. Pearce and R. E. Roth, "A G-20 Program for Studying Xenon Spatial Oscillations and Application to CANDU," CRRP-1151, July 1963

reduced. Solution of this problem requires a 3-dimensional analysis. An investigation into possible approaches to solution of this aspect of the xenon problem was initiated. Work using modal methods, currently in process at Oak Ridge National Laboratory, is being reviewed and possible three-dimensional solutions to the diffusion equation using other approximation techniques are being investigated.

2. Development Tests

About 90% of the components required for the single rod test device have been fabricated. Assembly of the rack and pinion subassembly has been initiated.

A ball bearing, made of 440-C stainless steel, has been under test in boiling water for 3 months. No visible corrosion is evident and the test is continuing.

3. Power and Xenon Control Rod Actuator Reliability Analysis

A reliability analysis for the power and xenon control system was prepared. The analysis shows that a 10-rod power control system has a reliability of 0.9639 with 2 permissible failures for a 30-yr lifetime. To demonstrate a system reliability of 0.9639 it will be necessary to subject each of 9 drive mechanisms (3 tri-foil assemblies) tested through 1.13 lifetimes with zero failures. Such a test program is based on achievement of a 50% confidence level.

D. SOLUBLE POISON CONTROL SYSTEM

A review was made of various neutron absorber materials soluble in D_2O . A soluble neutron absorber is being used in the HWOCR as a means of holding down the excess reactivity normally held by equilibrium xenon; this reactivity amounts to about 2.75%. The following absorber materials were studied: He^3 , gadolinium, cadmium, boron, and lithium. Of these materials the two most suitable for use in the HWOCR were boron in the form of boric acid and gadolinium in the form of gadolinium nitrate.

Gadolinium nitrate appeared to have an advantage over boric acid because of its ease of removal with conventional reactor grade mixed bed resins. However, very little operating data are available on the long-term effects of this material in a reactor loop.

Boric acid has been and is being used in water reactors without causing any significant corrosion or radioactivity effects. The burnout half life of about 10 days in a HWOCR flux of 2×10^{14} n/cm² sec limits the rate of reactivity increase due to burnout such that core reactivity during operation can be readily controlled by external systems such as the power control rod system and the boron injection and removal system.

As a result of this review, boric acid has been selected as the primary soluble neutron absorber for holding down the reactivity associated with equilibrium xenon in the HWOCR. Gadolinium nitrate has been chosen as the soluble poison to be used in the back-up shutdown system in the initial design concept.

III. NEXT REPORT PERIOD ACTIVITIES

The following tasks are planned for the next quarter.

- a) Complete a 2000-cycle test on the shutdown test device to determine wear conditions.
- b) Complete design of a reference shutdown test device based on previous test data.
- c) Study dynamic behavior of the shutdown test device using photographic and high speed data recorder techniques.
- d) Continue compatibility tests of bearing materials.
- e) Initiate testing of the single power and xenon control rod actuator test device.
- f) Initiate component testing of several segmented absorber elements.
- g) Review alternate methods of reactivity control to override xenon.
- h) Issue topical report on review of absorbers soluble in D₂O.
- i) Initiate specification for the equilibrium xenon control system requirements and for the emergency shutdown control requirements based on soluble absorber review.

HWO CR QUARTERLY REPORT

TITLE: Fuel

WBS NO: 28000

REPORTING PERIOD: October through December 1966

I. PROJECT OBJECTIVES

The objective of the fuel development program is to provide a fuel element system which meets the HWO CR requirements. This program includes design, fabrication, development testing, and evaluation of fuel element assemblies which demonstrate high neutron economy and low fuel cycle costs.

The present HWO CR reference fuel string consists of rod bundle type assemblies using hyperstoichiometric uranium carbide clad with SAP.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. FUEL MATERIALS

No work was performed on this program during the report period. Efforts on this program are deferred until FY 1968.

B. SAP CLADDING DEVELOPMENT

The objectives of the SAP development program are to:

- 1) Develop reliable sources for SAP alloys; evaluate and establish quality and uniformity of SAP raw materials.
- 2) Develop fabrication processes and product specifications for SAP cladding.
- 3) Develop mechanical properties data on SAP alloys directly applicable to HWO CR design.
- 4) Develop nondestructive and destructive inspection techniques for raw materials and cladding.
- 5) Maintain technical cognizance of ORNL SAP development program and other programs relating to the application of SAP to HWO CR.

AI-CE-56
WBS 28000

TABLE 28000-1
MONTECATINI BILLET RECEIVALS AND SCHEDULE

Alloy	Diameter (in.)	Amount (lb)	Scheduled Completion Date	Status	
				Material	Supplier QC Data
SAP 895	7	2500	6-30-66	Received	Received
SAP 895	7	500	7-31-66	Received	Received
SAP 895	2	1800	7-31-66	Received	Received
SAP 895	7	2200	8-31-66	In transit	Received
SAP 930	2	680	9-30-66	Received	Received
SAP 930	2	1530	9-30-66	In transit	Received
SAP 930	2	400	10-31-66	In transit	Received
SAP 930	7 (solid)	300	10-31-66*		
SAP 895	7 (solid)	2100	10-31-66*		

*This part of the order is subject to change because of difficulties in producing solid 7-in. diameter billets.

TABLE 28000-2
ALCOA XAP 005 BILLET RECEIVALS AND SCHEDULE

Diameter (in.)	Amount (lb)	Scheduled Completion Date	Status	
			Material	Supplier QC Data
2	2100	5-6-66	Received	Received
7	1200	8-3-66	Received	In-test
7	6500	10-4-66	Received	In-test
2	6400	1-31-67		

1. SAP Procurement

Purchase orders for SAP billet stock were placed with Montecatini and Alcoa in December of 1965 for GFY 1967 and 1968 HWOCR Development program requirements. A total of 26,000 lb of material is being procured as billets for subsequent fabrication into finished shapes as required. Two diameter sizes of billets were ordered; 2-in. for cladding tubes and end closure material, and 7-in. for pressure tubes.

a. Montecatini, Milan, Italy

The production and shipment of SAP 895 and SAP 930 billet material has proceeded normally during the quarter except for the production of solid 7-in. diameter SAP 895 billets. The supplier has reported that experimental extrusions have failed to yield good quality billets in the lengths required for HWOCR pressure tube fabrication. Montecatini has asked for a revision to the purchase order, which would permit substitution of hollow 7-in. diameter material for the quantity of solid 7-in. diameter billets presently specified. A review of HWOCR program requirements for SAP is being performed to determine the appropriate course of action.

A total of 7000 lb of SAP 895 billets has been fabricated as indicated in Table 28000-1. All of the 7-in. billets fabricated have 2.6-in. diameter holes through their centers. In addition, 2600 lb of SAP 930 2-in. diameter billets have been fabricated. A procurement status of the SAP 895 and SAP 930 is summarized in Table 28000-1.

b. Alcoa, Pittsburgh, Pa.

In accordance with agreements reached with Alcoa, the purchase order and specification were revised. The changes involved the metallographic and macroscopic examination, as well as the sampling plan, to be used on this order.

A total of 9800 lb of XAP 005 billets was received. This total includes 2100 lb of 2-in. diameter billets and 7700 lb of 7-in. diameter solid billets. A procurement status for XAP 005 billets is presented in Table 28000-2.

2. SAP Billet Evaluation

Work was initiated to characterize the SAP billet stock being received this fiscal year. The characterization studies are in addition to the quality control

testing performed by the suppliers and by Atomics International for acceptance purposes. The studies include the following: (1) tensile strength, yield strength, and elongation at room temperature and at 900°F; (2) Al_2O_3 content and homogeneity; (3) impurity elements; (4) microstructure; (5) macrostructure; and (6) internal soundness. These characterization studies are being performed to provide information on billet variability and for acceptability in development work.

Those billets currently being evaluated are the 2- and 7-in. diameter Alcoa XAP 005, and the Montecatini 2- and 7-in. diameter SAP 895. Generally, all testing is completed except for analytical chemistry and some microstructural studies. The following descriptions summarize the test results completed during this report period.

Aluminum oxide analyses are in progress on randomly selected billets. The analyses are being performed by the fast neutron activation (FNA) method. The available data on Al_2O_3 content are shown in Table 28000-3 for the respective alloys and billet sizes.

TABLE 28000-3
ALUMINUM OXIDE CONTENT OF ALLOYS AND BILLETS

Alloy	Billet Diameter (in.)	Number of Billets Sampled	Specified Al_2O_3 (wt %)	Al_2O_3 Analyses (wt %)
XAP 005	2	8	5.0 - 8.0	5.4 - 6.6
	7	5	5.0 - 8.0	5.5 - 6.4
SAP 895	2	2	9.0 - 12.0	10.6 - 11.0
	7	4	9.0 - 12.0	10.2 - 10.4

The impurity contents determined for these same billets are shown in Table 28000-4. Impurity analyses are in progress on additional billets of each alloy.

Inclusion counts were made on the transverse plane of 1-in.² samples from selected billets. The results, shown in Table 28000-5, indicate that the

TABLE 28000-4

IMPURITY COMPOSITIONS OF SAP 895 AND XAP 005 BILLETS
(ppm)

Element	Maximum Level*	2-in. Diameter Billet		7-in. Diameter Billet	
		SAP 895	XAP 005	SAP 895	XAP 005
Boron	3	<3	<3	<3	<3
Calcium	100	<100	<100	<100	<100
Cadmium	30	<10	<30	<10	<10
Carbon	8000	2250 - 2700	3700 - 7100	5750 - 6450	4500 - 7500
Copper	100	<20	<20	<20	<20
Gallium	300	<40	50 - 75	50 - 75	<200
Iron	2000	<2000	<2000	<2000	<2000
Molybdenum	50	<10	<10	<10	<10
Lead	40	<40	<40	<40	<10
Silicon	1000	<1000	<1000	<1000	<1000
Sodium	100	<100	<100	<100	<100
Tin	300	<50	<50	<50	<50
Zinc	500	<130	-	<100	<130
Zirconium	100	<50	<100	<50	<50
Nitrogen	800	380 - 990	132 - 1038	130 - 400	132 - 248
Hydrogen	20	10 - 17	10 - 18	16 - 18	9 - 17

*Maximum impurity levels are defined by Specification STO 170-NB 0042, Rev. A for Alcoa's XAP 005 and by Specification STO 170-NB 0041, Rev. A, for Montecatini's SAP 895.

TABLE 28000-5

INCLUSION COUNT DATA ON BILLETS

Alloy	Billet Diameter (in.)	Number of Billets Sampled	Average Number of Inclusions/in. ² of Surface		
			0.0005 to 0.001 in.	0.001 to 0.003 in.	>0.003 in.
XAP 005	2	6	12	18	16
	7	4	14	14	3
SAP 895	2	6	4	0	0
	7	10	2	0	0

Montecatini material is significantly more free of inclusions than the Alcoa material. Photomicrographs, in Figures 28000-1, 28000-2, and 28000-3, show typical inclusions found in the XAP 005 billets.

Macrostructure studies were performed on full transverse samples from the randomly selected billets. Samples from the billets were etched with 10 wt % NaOH for 10 sec at room temperature. No discontinuities, segregation, or structure abnormalities were noted for the 2-in. XAP 005 billets. Crack-like discontinuities were found in two out of four of the XAP 005 7-in. billets examined. Discontinuities in one of these billets are shown in Figure 28000-4. Of the SAP 895 billets examined, only one from 22 billets revealed any discontinuity. This was a 2-in. billet and the discontinuity is shown in Figure 28000-5.

The ultrasonic test results for six of the 2-in. diameter SAP 895 billets are given in Table 28000-6. Each of the 18-ft-long billets were cut into six 3-ft lengths, as indicated by the dash number following the three digit billet number. These data indicate a substantially cleaner condition for the 2-in. Montecatini billet, as compared to the 2-in. Alcoa billet. The ultrasonic testing was performed using the immersion technique and a formed lithium sulphate transducer. The instrument was set to indicate 100% of CRT range on a No. 5 Alcoa Area Amplitude Reference Block. Defect indications of 50% full scale were noted.

All thirty-seven 7-in. diameter SAP 895 billets from the Montecatini first shipment were ultrasonically tested. Twenty-nine of these billets did not exhibit any indications greater than the acceptance standard. Table 28000-7 summarizes the indications from the remaining eight billets. Billets were tested over 100% of the circumferential area by an immersion technique using the following procedure:

Scan speed — 25 rpm

Scan index — 0.030 in.

Transducer — 3/8-in.-diameter 10 MH, flat, lithium sulphate

Mechanical property data for the respective alloys and billet sizes are given in Table 28000-8.

Etch: 10 sec in 10 wt % NaOH at R. T.
Figure 28000-1. Metallic Inclusion in Alcoa 2-in.
Diameter XAP 005 Billet (Typical in 0.0010 to
0.0030 in. range. View is of transverse
section from front of billet No. 41-6)
(100X)

Etch: 10 sec in 10 wt % NaOH at R. T.
Figure 28000-2. Metallic Inclusion in Alcoa 2-in.
Diameter XAP 005 Billet (Typical in 0.0030 in.
range. View is of transverse section from
rear of billet No. 59-B)
(100X)

Etch: 10 sec in 10 wt % NaOH at R. T.
Figure 28000-3. Large (0.11 by 0.015 in.)
Metallic Inclusion in Sample from
Billet No. 45-1 (Alcoa 7-in.
diameter XAP 005)
(100X)

Etch: 10 sec in 10 wt % NaOH at R. T.
Figure 28000-4. Rear View of Discontinuities
on Cross Section of Billet No. 45-1
(Alcoa 7-in. diameter XAP 005. Actual size)

Figure 28000-5. Discontinuity in Rear End of
2-in. Diameter SAP 895 Montecatini Billet
No. 557. (Light areas are uneven
etch regions. Actual size.)

TABLE 28000-6
ULTRASONIC TEST OF 2-in. DIAMETER
MONTECATINI SAP 895 ALLOY BILLET

Billet Number	Number of Discontinuities/ft	Billet Number	Number of Discontinuities/ft
539-1	0	554-4	0
539-2	1	554-5	1
539-3	0	554-6	1
539-4	1	557-1	1
539-5	7	557-2	4
539-6	0	557-3	3
540-1	0	557-4	0
540-2	1	557-5	1
540-3	0	557-6	1
540-4	0	561-1	0
540-5	0	561-2	2
540-6	0	561-3	3
553-1	1	561-4	1
553-2	1	561-5	0
553-3	0	561-6	0
553-4	0	563-1	1
553-5	1	563-2	2
553-6	0	563-3	0
554-1	0	563-4	1
554-2	1	563-5	0
554-3	0	563-6	1

Exhibiting 20 to 100% of saturation CRT signal amplitude at test sensitivity

TABLE 28000-7
ULTRASONIC TEST OF 7-in. DIAMETER
MONTECATINI SAP 895 ALLOY BILLETS

Billet Number	Number of Discontinuities	Depth From Surface (in)	Range of CRT Signal Amplitudes	
			Sensitivity Level 1 (%)	Sensitivity Level 2† (%)
501	6	1/8 - 1/4	100	75 - 100
502	9	1/8 - 5/16	60 - 100	40 - 100
503	1	3/8	75	40
506	1	3/8	80	50
511	1	1/4	100	100
513	1	-	-	-
520	1	1/8	100	90
528	1	7/16	90	70

Instrument set for 100% CRT range on No 3 Alcoa Reference Block

†Instrument set for 100% CRT range on No 5 Alcoa Reference Block

AI-CE-56
WBS 28000

TABLE 28000-8
MECHANICAL PROPERTIES DATA OF ALLOYS AND BILLETS

Alloy	Test Temperature (°F)	Number of Tests	Ultimate Tensile Strength (ksi)	Yield Strength (ksi)	Elongation (%)
Transverse Mechanical Properties - 7-in -Diameter Billets					
XAP 005	RT	4	33 6 - 37 4	26 4 - 28 6	11 0 - 17 0
SAP 895	RT	10	41 1 - 44 7	26 3 - 29 3	7 0 - 8 0
XAP 005	900	4	7 8 - 9 3	7 5 - 9 1	3 0 - 4 0
SAP 895	900	10	8 7 - 10 8	7 3 - 9 5	3 0 - 4 0
Longitudinal Mechanical Properties - 2-in -Diameter Billets					
XAP 005	RT	3	33 6 - 35 7	27 5 - 28 0	20 0 - 22 0
SAP 895	RT	11	40 7 - 44 6	27 3 - 30 8	12 0 - 14 0
XAP 005	900	3	8 7 - 9 1	7 9 - 8 3	2 5 - 3 0
SAP 895	900	11	11 7 - 12 9	10 5 - 12 3	4 0 - 8 0

TABLE 28000-9
TUBE BURST STRESS RUPTURE TESTS OF CLADDING

Alloy	Test Temperature (°F)	Specimen Number	Hoop Stress (psi)	Rupture Time (hr)	In Progress Time (hr)
XAP 001	750	25B	6430	3288	
XAP 001	750	134A	6430	35	
XAP 001	750	135A	6075	35†	
XAP 001	850	17A	4400	1976	
XAP 001	750	133B	6430	1	
XAP 001	750	103G	6430	15	
XAP 001	900	47F	3800	519	
XAP 001	750	200B	6075	3319 ^a	
XAP 001	850	230A	4650	999	
XAP 001	850	90D	4650	926	
XAP 001	750	201C	6430	1256	
XAP 001	750	706G	6430	934	
XAP 005	750	357	5450	35†	
XAP 005	850	310	4400	371	
XAP 005	900	355	3100	875	
XAP 005	900	311	3500	639	
XAP 005	850	310A	4100	2704	
XAP 005	850	232A	4650	1608	
XAP 005	900	315	3500	2170†	
SAP 895	750	401B	6710	3239†	
SAP 895	750	403	6940	539†	
SAP 895	900	20	4800	395	
SAP 930	850	67G	4650	14	
SAP 930	900	29A	3800	303	
SAP 930	850	50K	4650	539	
SAP 930	900	71A	3800	748†	
SAP 930	850	83G	4400	3532	
SAP 930	900	91C	3450	1215	
XAP 001	900	72B	3450		1656
XAP 001	900	229A	3800		2256
XAP 001	750	300A	6075		5204
XAP 001	750	213C	6075		1920
XAP 005	750	315A	5750		2256

Specimens developed slow leaks Failure of braze joint is possible Specimens still in furnace and will be examined upon removal

†Specimens have not been examined because they are in retorts with other specimens still on test Data for these specimens are therefore tentative pending examination

3. Tube Stress-Rupture Tests

Biaxial stress-rupture testing of finned cladding tubes has continued on a few low-stress specimens from material used in the U^{305} irradiation experiments. Data accumulated to date are being statistically analyzed. Continuation of tube burst stress-rupture testing will use specimens representative of current production of Montecatini and Alcoa materials.

Data accumulated since the last report (AI-CE-55) are shown in Table 28000-9.

C. FUEL ASSEMBLY FABRICATION DEVELOPMENT

1. Magnetic Force Welding (MFW) Machine

a. Modifications

The MFW machine was relocated in the UC Pilot Fuel Production Facility at Santa Susana. Some malfunctions were encountered during checkout and were corrected. Concurrent with fabrication of closures for MFW development, the vacuum system and glove box for the welder were modified. Piping necessary to provide helium purge and atmosphere in the glove box was installed. Vacuum leaks in the vacuum chamber were located and rework done to eliminate them. At the end of the report period, the entire vacuum-helium system was ready for checkout.

The moving and reconnection of the MFW machine required that welds be made to verify the operational characteristics of the machine. A series of welds made for Piqua fuel rods prior to the move were repeated and evaluated as described below under Section b.

b. Joint Development and Testing

Welding development on Piqua fuel rods has been performed on both the large and small diameter rods. Tensile tests at 750°F on the small Piqua MFW closures were completed. The last EVOP* weld series on the large Piqua MFW closures were also made and 750°F tensile tests were completed. Statistical evaluations of the data indicate that the highest strengths were obtained by using the welding parameters shown in Table 28000-10. The average and range of ultimate tensile strengths are shown in Table 28000-11, for the closures fabricated

*Evolutionary Operation — A statistical approach to the evaluation of process parameters — used here to optimize magnetic force welding parameters.

TABLE 28000-10
MFW PARAMETERS FOR PIQUA FUEL
END CLOSURES

Cladding Cross Section Area*	Small Tube (0.034 in. ²)	Large Tube (0.045 in. ²)
Preload (lb)	800	600
Magnet Gap (in.)	0.160	0.135
Magnet Current (amp)	38,000	38,000
Weld Current (amp)	44,000	58,000
Clapper Size (in.)	0.25	1.5
Phase Shift (%)	10	10

*Nominal cross-sectional areas for the small and large Piqua cladding are assumed.

TABLE 28000-11
TENSILE STRENGTH OF PIQUA MFW END
CLOSURES AT 850°F

	No. Tested	Ultimate Tensile Strength (psi)		
		Average Strength (psi)	Highest Tensile Strength	
			Maximum	Minimum
Large Piqua*	4	10,266	10,666	9,822
Small Piqua†	4	8,852	9,324	8,471

*All tensile failures occurred partially or wholly in tube.

†One-half of tensile failures occurred partially in tube.

using these parameters. Of the large Piqua MFW closures tested, all broke at least partially in the parent tube. About half of the small Piqua closures broke partially in the parent tube; the other half broke through the weld. As indicated by the data in Table 28000-11, the scatter in breaking stress for both closures is quite narrow.

A series of room temperature and 850°F compression tests were made on Piqua cladding with and without MFW closure. Closures were made with clad fins removed for 3/8-in. from the weld. Results of these tests are given in Table 28000-12. These data indicate that the strength of the cladding is not adversely affected by the MFW closure.

TABLE 28000-12
COMPRESSION TESTS ON LARGE PIQUA CLADDING (XAP 005)
WITH AND WITHOUT MFW END CLOSURES

	No. Specimens	Tempera- ture (°F)	Compressive Strength at Ultimate Load (psi)
Cladding with Fins	2	RT	24,300
Removed 3/8 in. from Ends	1	850	9,400
Cladding with Fins	2	RT	27,100
Removed 3/8 in. from Ends and with MFW Closures	2	850	9,400

The desirability of changing the fuel rod design to permit fins to extend to a position immediately adjacent to the weld interface prompted a set of trials with new tooling. A new collet was designed to hold the cladding over the fin tips. This allows MFW closures to be made without removal of fins, which provides additional support for the cladding tube wall. The original collet and new collet designs are shown in Figure 28000-6. Several MFW closures were made using 10% Al₂O₃ SAP in the EXP-WR1-953 geometry. Three of the closures were tensile-tested at room temperature with the results tabulated below. Considering that the welding parameters were scaled from Piqua settings, the strengths were exceptionally good.

*WELD FLASH HAD TO BE REMOVED IN ORDER TO CLEAR GRIP AREA. THIS METHOD WOULD BE VERY TIME CONSUMING WHEN MAKING THE SECOND HELIUM PURGED END CLOSURE.

2-8-67 UNC

7680-25971

Figure 28000-6. Original and New Collet Designs

<u>Specimen No.</u>	<u>Ultimate Load (lb)</u>	<u>Ultimate Tensile Strength (psi)</u>	<u>Location of Break (% tube break)</u>
23/29	1,055	30,740	80
25/28	975	28,620	95
26/27	1,055	30,740	95

Attempts were made at measuring interface temperature and forging ram response time during MFW closure fabrication. Theoretically, the highest strength joint should be obtained when peak temperature and forging force are synchronized. The data obtained indicate that maximum forging ram displacement is reached about 4 msec after peak interface temperature. Tests will be conducted to establish whether higher closure strengths can be obtained with peak values synchronized. This can be accomplished by firing the current to the magnet (ram movement) earlier than the welding current (interface heating).

2. Nondestructive Testing Development (NDT)

The feasibility of a miniaturized high resolution ultrasonic test for the MFW closure has been reported previously.* The successful application of this test was demonstrated for the "recessed fin" end closure design wherein the fins on the cladding were removed about 3/8 in. from the weld joint. However, the probable requirement for the cladding fins to extend up to the weld joint has resulted in the need to reestablish the feasibility of the ultrasonic test method. A study is necessary to determine the magnitude of the fin shadow behind which no flow detection can occur, and to determine what methods can be used to minimize or eliminate this shadow. This work is in progress and must be completed prior to procuring the equipment required for production applications.

3. End Plate Attachment

Effort for developing alternative schemes for attaching end plugs to end plates was continued. Fifteen more end cap specimens were cold-headed to make the mechanical joint to the end cap. When the previous batch was fabricated, the riveting machine manufacturer (H. P. Townsend Co.) recommended reduction of the included angle of the countersink, and the recent batch thus used a countersink with an included angle of 60° instead of the original 82°.

*Op. Cit. AI-CE Memo 55

Figure 28000-7. Components for Cold-Headed End Plug, Joint Trials

Figure 28000-8. Cross Section of Cold-Headed End Plug (5X)

The countersunk surface in the end plate was serrated by a hammer blow on the end of a recessed head screwdriver. The SAP material of the end caps was supposed to flow into the grooves and prevent rotation of the tubes. Some flow did take place but it was insufficient to prevent breaking the end cap loose from the end plate by hand and to prevent rotation.

Figure 28000-7 shows the two pieces (end cap and end plate) before and after joining. The grooves in the countersunk surface of the end plate are visible. Figure 28000-8 is a 5X enlargement of a metallographic section of one of the joined specimens. The end cap tip was slightly longer than necessary before forming, and there is excess material above the end plate. Notice that the SAP material did not completely fill the conical countersink. It is believed that this condition could be rectified, if necessary, by a small modification to the countersink. Tensile tests to determine the load required to pull the mechanical joint apart have not as yet been completed.

D. IRRADIATION TESTING

The objectives of this activity are to obtain irradiation data on the behavior of fuel and cladding materials and to evaluate the performance capabilities of HWO CR fuel element assemblies.

1. NRU Irradiations

a. U-305 Post-Irradiation Examination

Upon completion of the screening examination of Bundle LTA during the previous report period,* Rods B, H, J, and P of Bundle LTA were selected for the detailed fuel and cladding examinations. Rods B, H, and J were disassembled by removing the end caps and slitting the cladding into two halves by elox machining. The fuel and inner surfaces of these rods were then exposed for examination. Rod P was disassembled by sectioning into 1-in. wafers from several locations. The wafers were allocated for cladding ID measurements, fuel and cladding metallography, and fuel cladding compatibility studies. Subsequently, the cladding was removed from the remaining sections of Rod P by elox machining, and the fuel and cladding were subjected to detailed examination.

*Op. Cit. AI-CE-55

Kollmorgen composites were made of the fuel and of the inner surfaces of the cladding, and higher magnification stereo photographs were taken of selected areas. Figures 28000-9, -10, and -11 show the fuel and cladding from Rods B, H, and J, respectively and illustrate the excellent condition of the fuel and cladding. The fuel is relatively intact and the individual slugs are easily discernible. The cracks seen in the figures occurred during separation of the fuel and cladding. The inner surfaces of the cladding exhibited stains and fuel-like deposits (the white or silver dots seen in the above figures). The stains appear to correspond to the interfaces of the individual fuel slugs and to a surface cracking pattern observed on the fuel. This surface cracking pattern is characteristic of a stress-relieved condition in a cast rod. Some small fuel pieces were also observed on the cladding at the interfaces between slugs, but these are not adherent to the cladding and are thought to be the result of fuel shifting during handling and not an in-pile condition. The small spherical deposits (see the figures) are tightly adherent to the cladding and may be the result of minor fuel-cladding interaction.

Upon completion of the visual examinations, the diameters of the fuel from rods B, H, and J were measured at 1-in. intervals at 0 and 90° orientations. These measurements are tabulated by rod and slug identification in Tables 28000-13, -14, and -15. The measurements indicate that there was insufficient fuel growth to have uniformly contacted the cladding except in the area of end peaking of the top slugs. All fuel cladding contact was therefore the result of random contact and rod bow. This is further supported by the observation that most of the deposits seen on the cladding occurred on the 270° half of the cladding, i. e., on the concave side of the bow.

The density of the fuel was also measured. Comparison of density and dimensional measurements indicates that fuel diameters and lengths (where available) increased more than would be predicted by density measurements. This difference is explained on the basis of the open surface void volume due to surface irregularity and surface cracking.

Fission gas samples were taken from each of the rods of bundle LTA. The amount of gas released from the fuel (percent of gas theoretically produced) was low. The maximum collected from the outer rods was 0.032% from Rod J, and, from the inner rods, 0.008% from Rod Y based on the estimated burnup for the fuel.

2-10-67 UNC

7680-25974

Figure 28000-9. U-305 Bundle LTA Rod B Typical Fuel
and Cladding Appearance

2-10-67 UNC

7680-25975

Figure 28000-10. U-305 Bundle LTA Rod H Typical Fuel
and Cladding Appearance

2-10-67 UNC

7680-25976

Figure 28000-11. U-305 Bundle LTA Rod J Typical Fuel and Cladding Appearance

AI-CE-56
WBS 28000

TABLE 28000-13

B ROD DENSITY AND DIMENSIONAL DATA

Slug	C (wt %)	Diameter (0°)	$\Delta D/D_i$	Diameter (90°)	$\Delta D/D_i$	ρ_i	ρ_f	$\Delta\rho/\rho_f$
B11-7A	4.90	0.5033	0.86	0.5035	0.90	13.52	13.27	1.88
		0.5026	0.72	0.5030	0.80			
		0.5027	0.74	0.5021	0.62			
		0.5013	0.46	0.5023	0.066			
		0.5022	0.64	0.5025	0.70	13.52	13.30	1.65
		0.5021	0.62	0.5022	0.64			
		0.5018	0.56	0.5024	0.68			
		0.5022	0.64	0.5022	0.64			
B11-27A	4.88	0.5015	0.50	0.5020	0.60	13.51	13.28	1.73
		0.5020	0.60	0.5027	0.74			
		0.5023	0.66	0.5021	0.62			
		0.5025	0.70	0.5022	0.64			
		0.5017	0.54	0.5020	0.60	13.51	13.28	
		0.5017	0.54	0.5020	0.60			
		0.5015	0.50	0.5020	0.60			
		0.5020	0.60	0.5018	0.56			
B11-10A	4.85	0.5024	0.68	0.5022	0.64	13.52	13.29	
		0.5024	0.68	0.5025	0.70			
		0.5022	0.64	0.5024	0.68			
		0.5023	0.66	0.5026	0.72			
		0.5017	0.54	0.5018	0.56		13.27	
		0.5024	0.68	0.5022	0.64			
		0.5025	0.70	0.5027	0.74			
		0.5024	0.68	0.5026	0.72			
B11-7	4.90	0.5015	0.50	0.5024	0.68	13.53	13.27	1.95
		0.5018	0.56	0.5027	0.74			
		0.5024	0.68	0.5025	0.70			
		0.5014	0.48	0.5024	0.68			
B11-29	4.87	0.5021	0.62	0.5017	0.54	13.53	13.32	1.57
		0.5020	0.60	0.5018	0.56			
		0.5016	0.52	0.5023	0.66			

TABLE 28000-14
H ROD DENSITY AND DIMENSIONAL DATA

Slug	C (wt %)	Diameter (0°)	$\Delta D/D_i$	Diameter (90°)	$\Delta D/D_i$	ρ_i	ρ_f	$\Delta\rho/\rho_f$
H-7-9	4.97	0.5018	0.56	0.5014	0.48	13.51	13.29	1.66
		0.5017	0.54	0.5010	0.40			
		0.5020	0.60	0.5023	0.66			
H-7-23	4.95	0.5018	0.56	0.5013	0.46	13.51	13.28	1.73
		0.5023	0.66	0.5023	0.66			
		0.5022	0.64	0.5022	0.64			
		0.5024	0.68	0.5023	0.66		13.27	1.81
		0.5021	0.62	0.5021	0.62			
H-7-29	4.98	0.5021	0.62	0.5022	0.64	13.50		
		0.5021	0.62	0.5026	0.72			
		0.5023	0.66	0.5026	0.72			
		0.5018	0.56	0.5022	0.64		13.27	1.73
		0.5021	0.62	0.5023	0.66			
		0.5024	0.68	0.5027	0.74			
H-7-26	4.93	0.5020	0.60	0.5023	0.66	13.51	13.29	1.66
		0.5018	0.56	0.5018	0.56			
		0.5023	0.66	0.5019	0.58			
		0.5019	0.58	0.5013	0.46		13.27	1.81
		0.5022	0.64	0.5008	0.36			
		0.5017	0.54	0.5020	0.60			
H-7-18	4.94	0.5032	0.84	0.5044	1.08	13.51	13.24	2.04
		0.5030	0.80	0.5043	1.06			
		0.5033	0.86	0.5031	0.82			
		0.5012	0.44	0.5015	0.50		13.26	1.88
		0.5010	0.40	0.5008	0.36			
		0.5025	0.70	0.5024	0.68			
		0.5021	0.62	0.5023	0.66		13.28	1.73

TABLE 28000-15
J ROD DENSITY AND DIMENSIONAL DATA

Slug	C (wt %)	Diameter (0°)	$\Delta D/D_i$	Diameter (90°)	$\Delta D/D_i$	ρ_i	ρ_f	$\Delta\rho/\rho_f$
J-6-8A	5.01	0.5047	1.14	0.5035	0.90	13.49	13.20	2.19
		0.5041	1.02	0.5032	0.84			
		0.5038	0.96	0.5034	0.88		13.25	1.81
		0.5042	1.04	0.5031	0.82			
		0.5018	0.56	0.5033	0.86			
		0.5024	0.68	0.5028	0.76			
		0.5022	0.64	0.5026	0.72		13.25	1.81
J-6-1A	5.04	0.5021	0.62	0.5026	0.72	13.43	13.18	1.89
		0.5018	0.56	0.5027	0.74			
		0.5013	0.46	0.5022	0.64			
		0.5016	0.52	0.5020	0.60			
		0.5022	0.64	0.5023	0.66		13.19	
		0.5026	0.72	0.5022	0.64			
J-6-8B	5.01	0.5039	0.98	0.5029	0.78	13.45		
		0.5030	0.80	0.5031	0.82			
		0.5037	0.94	0.5035	0.90		13.22	1.73
		0.5030	0.80	0.5036	0.92			
		0.5037	0.94	0.5032	0.84		13.23	1.66
		0.5033	0.86	0.5038	0.96			
J-6-20B	5.08	0.5024	0.68	0.5027	0.74	13.40		
		0.5027	0.74	0.5022	0.64			
		0.5028	0.76	0.5036	0.92			
		0.5033	0.86	0.5031	0.82		13.18	1.66
		0.5030	0.80	0.5030	0.80			
		0.5032	0.84	0.5028	0.76			
J-6-6C	5.05	0.5024	0.68	0.5031	0.82	13.41	13.18	1.74
		0.5030	0.80	0.5034	0.88			
		0.5028	0.76	0.5032	0.84			
		0.5033	0.86	0.5035	0.90			

Cladding metallography was completed on all but one of the samples cut from Rods B, H, and J. A discontinuous apparent fuel/cladding interaction was observed on all three rods but was somewhat more extensive on Rod H than on Rods B and J. The reaction product appeared to be an intermetallic compound of high aluminum content with an average thickness of 0.5-mil and a maximum thickness of 1.0-mil. A typical example is shown in Figure 28000-12. The reaction appeared to extend from the fuel/cladding interface into the fuel except in areas of high mechanical interference. In these cases, the fuel apparently contacted the cladding and filled the surface irregularities. The fuel particles surrounded by the reaction product did not exhibit the expected etching response, indicating some deviation from the normal UC material.

Figure 28000-12. U-305 Bundle LTA Rod B Typical Metallography (250X etched) (Reaction is into fuel approximately 1 mil – not into cladding)

To summarize, the screening examination and the available information on the detailed examination of bundle LTA and selected fuel rods from that bundle continue to support the conclusion that the U-305 experiment was successful.

Examination of fuel and cladding showed the fuel to be relatively intact and revealed no cladding failures or significant defects. Fuel growth was not sufficient to contact the cladding except at rod ends which were affected by flux peaking.

A fuel/cladding interaction was observed and was studied metallographically. Metallography has shown some minor reaction zones on the cladding. Most of the reaction product (apparently an intermetallic of high aluminum content) appears to extend into the fuel rather than into the cladding. The maximum penetration into the cladding observed to date is approximately 1-mil.

b. EXP-NRU-305-A1 Operation

Operation of the U-305-A1 experiment was severely limited as a result of an extensive reactor shutdown from September 19 to October 24 and the temporary removal of the experiment from the reactor from November 14 to November 28. As a result, the experiment operated only 41 of the 92 calendar days in the report period, accumulating an estimated 1825 Mwd/MTU burnup.

An organic leak which developed in the graphite thermocouple seal was reduced to a level of about 30 cc/hr by tightening the seal. Further tightening was not attempted because of the possibility of shearing all the thermocouple leads.

The leak increased during the period from November 12 to November 13 and the experiment was removed from the reactor on November 15 to replace the graphite seal with a brazed seal and preclude a complete seal failure.

After installation of the brazed seal, the experiment was reinserted in the U-3 loop and, on heatup, another leak of about 1600 cc/hr was found. Replacement of a gasket between the test string and pressure tube assembly reduced this leak to between 20 and 30 cc/hr, an acceptable leak rate.

While the experiment was out of the reactor, it was observed that the zirconium hydriding specimens attached to the upper support tube were partially damaged. Consequently all of the specimens were removed from the experiment prior to reinsertion.

The seal replacement and associated thermocouple splices were successful in restoring to serviceability 22 of the 24 installed thermocouples. Between the time of string initial insertion and November 15, the number of useful thermocouples had decreased to 11.

During the extended shutdown of September 19 to October 22, two new ΔP cells were installed in the loop to augment the data from the original ΔP cell designated DP-2. These two new differential pressure transmitters are designated DP-81 and DP-82. The take-off points for these new cells were at different locations than those for DP-2 and differed from each other. Because it was desirable to compare the reading of the three cells on a common base to obtain useful data, corrections were determined to refer the new cell readings to the base reading of the DP-2 cell. These corrections were originally determined to be +8 psi for DP-81 and -4 psi for DP-82, with normal flow conditions of 16.4 kg/sec coolant through the loop. The +8 psi correction for DP-81 was later revised to +2 psi when a 6 psi depression in the DP-81 channel recorder was observed.

During the November shutdown, DP-2 cell was removed from the system because it occupied the space required by a TV camera which will be used to observe a planned NRU calandria change.

Two new pressure transducers were also added to the system, near the outlet and inlet ends of the test section PT-79 and PT-80. The head correction to be applied to the difference of these two pressure transducers to determine test section pressure drop is -11 psi.

Flow-delta P tests were conducted during each of the two shutdowns. The data are compared in Figure 28000-13. Three points of data (PT80-PT79-11) for the October test are also plotted in this graph. DP-82 was not in service during the October tests and is therefore not presented for comparison.

During the November shutdown, additional flow-delta P tests were conducted to provide reference hydraulic information using the dummy fuel element which was installed in the U-3 loop while the brazed seal was being installed. The data from these tests are illustrated in Figure 28000-14, comparing the readings of DP's 2, 81, and 82.

Figure 28000-13. U-305-A1
Flow-Delta P Tests

Figure 28000-14. Flow-Delta P
Test on Dummy Bundle

Table 28000-16 shows the range of the U-305-A1 operating parameters for the report period.

TABLE 28000-16
RANGE OF EXPERIMENT PERFORMANCE
SEPTEMBER 15-DECEMBER 15, 1966

	High	Low	Notes
Flow (kg/sec)	16.7	15.4	
DP-2 (psi)	107.0	96.0	1
DP-81 (psi)	95.0	91.0	2
DP-82 (psi)	95.5	90.0	2
Maximum Cladding Temperature (°C)	442	418	
Maximum Film Formation Parameter	2.6	1.9	
Inlet Temperature (°C)	312	300	
Delta Temperature (°C)	45.0	38.4	
Coolant Water Content (ppm)	488	194	

Notes: 1. Period from September 15 to November 30, prior to the installation of the new DP cells.
2. Period beginning October 25; first readings during operation of newly installed instruments.

Additional effort was expended during this report period to determine the extent and causes of the anomalous increases in the hydraulic resistance of the U-305 and U-305-A1 experiments. To date, no conclusions have been reached but significant supplementary information was developed.

One mechanism examined was the increase in friction factor (f) which could be caused by buildup of a rough fouling film on the elements. The maximum estimated film buildup was calculated as ~ 1 mil based on an assumed conductivity of $0.5 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$. The increase in friction factor caused by a 1-mil film with a 1/2-mil roughness was calculated to be about 50%. This would result in a 15 psi increase in the total test section pressure drop.

The effect of coolant physical property changes was found to cause a maximum of ± 5 psi change in the pressure drop over the range of operating conditions experienced during the U-305 and U-305-A1 experiments. This does not include the effect of density changes which are corrected by the data logger and computer in the measurement of flow rate.

Measurement accuracy of the test section flow and pressure drop is the most probable explanation for the pressure drop anomalies. Since the pressure drop is proportional to the square of the flow, a 10% increase in flow can cause a 20% increase in the pressure drop. For the normal U-3 condition, this would mean that a 1 kg/sec increase in flow would cause a 12 psi increase in pressure drop. Based on the limited information since the completion of the instrumentation improvements, much more reliable system behavior and consistent loop hydraulic data appear practical. The overall hydraulic history of U-305-A1 to date is summarized in Figure 28000-15.

Because of the extensive reactor outages during the quarter, coupled with the loop instrumentation checkout and seal replacement activities, no major film formation tests were performed. However, at the close of the report period, a low water test was initiated. The principal thermal history of the U-305-A1 experiment to date is summarized in Figure 28000-16.

2. WR1 Irradiation - EXP-WR1-951 Operation

Near the end of the report period the WR1 was shut down due to an undesirably high fouling potential in the A loop (SPFT's of $11 \mu\text{gm/cm}^2\text{-hr}$). During the

Figure 28000-15. Overall Hydraulic History of U-305-A1

AI-CE-56
WBS 28000

Figure 28000-16 (Part 1). Thermal History of U-305-A1

196

AI-CE-56
WBS 28000

2-13-67 UNC

7680-25981-1

Figure 28000-16 (Part 2). Thermal History of U-305-A1

shutdown, 2 of the 4 clay columns being used on the B loop were returned to the A loop to provide additional cleanup capacity. When WR1 operation was resumed on October 3, the values of the film formation parameters for the EXP-WR1-951 test were the same as those prior to shutdown. On October 7, the reactor coolant inlet temperature was reduced to 285 from 335°F to operate the clay columns at a lower temperature. Following the reduction in bulk coolant temperature the film formation parameters followed predictions by increasing about 8%. (See Figure 28000-17.)

Operation continued through October until a cycle shutdown on October 30. During subsequent fuel movement operations, an organic spill occurred in the upper access space which soaked the upper part of the reactor and the experiment instrument connectors, and drained past the reactor into the lower access space. This spill resulted in a 21-day shutdown during November. Operation was resumed on November 21 and continued until the scheduled shutdown on December 24.

The operational history of EXP-WR1-951 is described in Figure 28000-17. The experiment operated very smoothly during the report period and the most significant occurrence was an indication of a decrease in heat transfer resistance on the hotter thermocouples. From October 7 to December 24, the film formation parameter for TC 5 decreased by about 18%. Although the cooler thermocouple did not show as pronounced a decrease, it is concluded that the film formation potential of the coolant was reduced at the lower temperature. However, this conclusion may be tempered by coolant chemistry data, prior and subsequent to the reduction in the coolant temperature, since the water content did not differ appreciably and the MST and SPFT information did not indicate marked improvement.

Hydraulic tests and thermocouple calibrations were conducted on the EXP-WR1-951 during the report period. Figure 28000-18 compares hydraulic tests made on June 5 and October 2. The test results indicate that there were no significant changes in the hydraulic characteristics of EXP-WR1-951.

A thermocouple calibration was made on November 29 during reactor heatup. The results of this calibration indicate that, for all the operating thermocouples, indicated values remain within the manufactured specification of $\pm 0.75\%$ or

Figure 28000-17. EXP-WR1-951 Operational History

AI-CE-56
WBS 28000

Figure 28000-18. EXP-WR1-951 Hydraulic Tests

$\pm 3.75^{\circ}\text{F}$ at 500°F . The performance of these thermocouples is excellent and to date only 6 out of the 33 thermocouples are not using the Chromel-Alumel junctions. Temperature signals are obtainable from most of these 6 thermocouples using a grounded circuit.

The operating status of EXP-WR1-951 as of December 23, 1966 is contained in Table 28000-17.

TABLE 28000-17
EXP-WR1-951 OPERATING STATUS

951 Power (Mwt)	1.106
Inlet Coolant Temperature ($^{\circ}\text{F}$)	542
Coolant Delta T ($^{\circ}\text{F}$)	44.5
Coolant Flow Rate (gpm)	304
Maximum Measured Cladding Temperature ($^{\circ}\text{F}$)	723
Accumulated Peak UC Burnup (Mwd/MTU)	4622

During the period, improvements were made to the data acquisition, computation and distribution system. At the irradiation site, the data logger was modified to collect data on inlet and outlet temperatures, flow rate and moderator level. This information, coupled with the existing fuel string temperatures, will allow computer handling of the punched paper tape to include computations of experiment power and film formation parameters. This expanded capability will increase the detail at which the experiment can be followed at the site. A condensed data file to be used for long term analysis and computing trends is also being assembled. Computer codes for preparing the punched paper tape data for analysis have been written.

3. Design and Fabrication of WR1 Test Fuel Elements

a. EXP-WR1-952

It was decided that the 952 fuel bundles will be banded with stainless steel bellybands. Effort was initiated on the design of the bands.

b. EXP-WR1-954

The 954 test assembly was almost completed by the end of the quarter. The scheduled shipping date was deferred due to the addition of the requirement for banding the fuel bundles (see EXP-WR1-952 above). The test procedure and test reports for 954 were deferred pending completion of the design, fabrication, and installation of the bellybands.

c. EXP-WR1-953

This test was deferred to allow funding of the design and fabrication of the NAA-84 test series (see SAP Irradiation - NAA-84 test series below).

d. EXP-WR1-956

The design of the 956 test was rescheduled for initiation late in FY 1967 due to conflicts with the design of the NAA-84 test series and the EXP-NRU-307 test.

4. NRU Irradiations

a. EXP-NRU-305A2

Examination of the U-305A2 string was completed at Chalk River. A detailed boroscopic examination of the interior flow channels on the end bundles

was made and surface finish measurements were taken. Four outer rods on each of the 6 bundles were radiographed to assess whether significant fuel breakup might have occurred when the spacer rings were deformed. The deposits were determined to be nondetrimental to the operation of the string. Surface finish measurements on the heat transfer surfaces showed little variation from corroded to noncorroded fuel rods (e.g., 45-80 μ -in. rms on corroded surfaces vs 27-76 μ -in. rms on noncorroded surfaces). However, the deposits on the silver-rich end plug regions were heavier than on the heat transfer surfaces. Boroscopic examination revealed the inner surfaces of the outer fuel rods were essentially the same in appearance as the outer surface. No channel plugging was observed.

Radiographic examination revealed one broken fuel slug in bundle LPA and an $\sim 1/2$ -in. separation of the slug from the adjacent slugs. Small end chips were also noted in two other peripheral rods in bundle LNA. These effects are not considered deleterious.

The spacer rings were straightened using a specially designed fixture. The spacer rings, which had measured diameters ranging from 3.165 to 3.253 in. prior to straightening, had measured diameters ranging from 3.208 to 3.222 in. after straightening, which is within fabrication tolerance.

b. EXP-NRU-307

The preliminary nuclear analysis of the U-307 test, including the determination of power vs burnup, was completed. Other preliminary analyses were deferred pending results of the out-of-pile wear tests being conducted at Combustion Engineering in the TF-10 loop. Because of the failure of TS 10-4 (Ref. Section 21000) the design of the U-307 will have to proceed without the benefit of comprehensive wear test results from TF-10.

Some of the design parameters have been established for the U-307 test. The major parameters remaining are final cladding dimensions (therefore final fuel dimensions) and banding requirements.

AI-CE-56
WBS 28000

2-10-67 UNC

7680-251007

Figure 28000-19. Knoop Hardness vs Location Across UC-SAP Interface, Capsule DH-70 (150X)

E. OUT-OF-PILE TESTING

1. SAP-UC Compatibility

The goals of the UC-SAP Compatibility Tests are to define the nature and extent of reaction between hyperstoichiometric UC fuel and SAP cladding. The investigation method being used is the metallurgical evaluation of UC-SAP interfaces to identify and measure the extent of reaction that occurs as a function of various time-temperature test conditions.

In the earlier qualitative screening evaluation of UC-SAP compatibility test specimens, three general categories were selected to identify the extent of the observed reactions. These categories were designated: "major," extensive and continuous formation of an intermetallic layer at the SAP-UC interface; "minor," a small but metallographically detectable region of intermetallic formation; and "no reaction," no formation of intermetallic material at the interface based on metallographic inspection. Additional studies during the current report period have yielded more detailed data in each of the three categories. These studies utilized a combination of (a) detailed metallographic examination of compatibility sample interfaces, (b) hardness traverses across the sample interfaces, and (c) electron microprobe measurements across the test sample interfaces. Typical examples of the results of hardness traverses and detailed metallographic examination are shown in Figures 28000-19, -20, -21, and -22.

Figure 28000-20. Gradient Microstructure Next to the "Major" Reaction of the 5.3 wt % C in UC-SAP Interface of Capsule DH-70 (150X)

2-10-67 UNC

7680-251008

2-10-67 UNC

7680-251009

Figure 28000-21. Knoop Hardness Location Across UC-SAP Interface, Capsule DH-87 (150X)

Figures 28000-19 and -20 show extensive formation of intermetallic material. The variations in hardness in the sample are noted by changes in the size of the indentations as a function of position. The Knoop hardness members are also presented in Figure 28000-19. The formation of an intermetallic zone ~10-mils thick is apparent from both the photomicrograph and the hardness measurements. In addition, it was found that some type of effect extends past the visible intermetallic zone into the SAP as indicated by the hardness measurements and the photomicrograph in Figure 28000-20. Electron microprobe measurements revealed the aluminum atom density in this region to be less than in the parent SAP material.

The sample that displayed massive formation of intermetallic reaction product may be contrasted with the samples illustrated in Figures 28000-21 and -22. These samples were heat-treated at lower temperatures than the sample in Figures 28000-19 and -20 (925 and 950 vs 1050°F). Only slight reaction is evident in Figure 19000-21 and no reaction product was found in the sample shown in Figure 19000-22. No evidence of an effect extending into the SAP was found either by hardness measurements or by metallographic methods.

In addition to the out-of-pile tests, samples of the fuel cladding from the NRU-305 test element were examined for evidence of a UC/SAP reaction. This element operated for the equivalent of 213 full power days (5100 hr) at an average fuel centerline temperature of 1900°F (2100°F peak). The average and peak XAP-001 cladding temperatures during operation were 850 and 870°F, respectively. The results of the examination are illustrated in Figures 28000-23, -24, and -25. In the sample shown in Figure 28000-23, a thin discontinuous layer of unidentified material (~0.001-in. thick) adhered to the inside surface of the SAP. This material appears to have a microstructure similar to that of the XAP 001 cladding. A possible explanation of this observation is that a tiny chip of cladding material was trapped between the fuel and cladding and became bonded to the cladding during irradiation. Figure 28000-24 shows a particle of hard material, that appears to be UC fuel, imbedded in the XAP 001 cladding material. Even this condition of intimate contact did not result in a significant reaction between the carbide and SAP. Figure 28000-25 illustrates particles of material that appear to be UC fuel adjacent to the XAP 001 cladding material. In this case, the fuel particles were trapped against the cladding by a material

2-10-67 UNC

7680-251010

Figure 28000-22. Knoop Hardness Location Across UC-SAP Interface,
Capsule DH-74 (150X)

UNC

28000-251011

Figure 28000-23. Material Adhering to Surface of XAP 001 Cladding from NRU-305 Irradiation Test (~0.001-in. thick; as polished) (200X)

UNC

7680-251012

Figure 28000-24. Small Spherical Fuel Particle (~0.002-in. diameter) Embedded in the Cladding from NRU-305 Irradiation Test (500X)

Figure 28000-25. Material Appended to SAP Cladding Surface with Included Particulate Matter (From NRU-305 Irradiation Test) (500X)

that also has a microstructure similar to that of the cladding. Again, no evidence of a significant reaction between the trapped particles and the surrounding material was observed.

A review of some samples from the out-of-pile compatibility tests which had been previously classified as "no reaction" was conducted at higher magnification (1000 vs 150X). Some samples contained very small isolated areas of reaction product. One type of condition that exhibited detectable evidence of intermetallic formation (approximately 1 to 2 microns thick) is illustrated in Figure 28000-26 where SAP material was actually extruded into a crack or void in the surface of the uranium carbide. In this area the protective aluminum oxide film was disrupted as a result of the plastic flow of the SAP, and aluminum contacted the uranium carbide directly. This condition is believed to be the consequence of very high contact pressures, far in excess of the contact pressure expected in HWO CR fuel elements, and is therefore not considered of engineering importance in the reactor. It does illustrate, however, that very slight reactions can occur at HWO CR fuel cladding temperatures under ideal conditions. Similar reactions on a micro scale were observed in a capsule heat treated at 925°F for 500 hr.

Figure 28000-26. Intrusion of SAP Spike into Hyperstoichiometric UC of Capsule DH-83. (Interface between SAP spike and UC shows reaction zone. Capsule heat-treated 2000-hr at 850°F) (1000X)

The observations made to date on UC-SAP compatibility are summarized in Figure 28000-27. It should be noted that no distinction was made between samples with very high contact pressures and those with contact pressures in the normal range expected in HWO CR fuel elements (800 to 1000 psi). The majority of the tests performed to date have been designed so that the contact pressures are substantially in excess of this range. Even so, the data indicate that, for the conditions of these tests, significant reactions are not expected at temperatures below about 925°F.

2. Hydraulic Testing

The HWO CR fuel bundle hydraulic testing program is designed to determine the hydraulic performance characteristics of typical HWO CR fuel bundles and associated hardware. The present objective is to determine the coolant mixing (adjacent channel cross flow) characteristics of the reference design close-pack array, 37-fuel-rod bundle. Velocity profile, pressure drop, and vibration characteristics data will be obtained from these tests, but are not primary objectives. The objective is being accomplished through testing of fuel

Figure 28000-27. Hyperstoichiometric UC/SAP Compatibility Test Data
(Capsules DH-1 through DH-88; status on 12-16-66)

bundle models which geometrically simulate all, and portions of, the cross section of the 37-fuel rod bundle.

The data from the bundle modular tests will be used to predict bundle performance for fuel bundle design verification and upgrading. Tests on a model of a complete fuel element bundle, under simulated reactor flow and temperature conditions, will be used to determine performance capability.

a. Full-Scale Coolant Mixing Tests

Coolant mixing tests were continued in the TF-3 facility. Additional instrumentation was located near the center of the mockup fuel bundles. The locations of the various instrumented subchannels are shown in Figure 28000-28. The tests, for both up and down flow, are being conducted in room temperature and 170°F water, with average bundle velocities ranging from 10 to 40 ft/sec. Dye distribution curves for the inner subchannels are in good agreement with anticipated results. A typical plot for injection into subchannel K is shown in Figure 28000-29.

Additional test data from the predicted hot channel A indicate a somewhat different mixing pattern for the peripheral subchannel C. A typical dye distribution plot for injection into subchannel A is presented in Figure 28000-30. Comparison of this curve with Figure 28000-16 in the previous quarterly,* indicates that the dye concentration in the peripheral subchannel C is considerably lower in this test. This is attributed to the fact that the loading on the bundle assemblies was less than that in the first series of tests, which probably decreased the bowing of the outer rods towards the pressure tube wall and increased the cross flow between the pressure tube wall and the rods. To confirm this assumption, the bundles will be subjected to the original loading and the tests repeated.

Preliminary analysis of the mixing data in the inner subchannels indicates the cross flow rates are ~0.50 lb/ft-sec, based on the density of water at 80°F, at an average fuel bundle velocity of 30 ft/sec.

b. Multichannel Cross Flow Test

The testing of the multichannel model was completed with the results showing good correlation with previous bundle flow tests. The average ratio of cross

*Op. Cit. AI-CE-55

Figure 28000-28 Instrumented Subchannel
Locations

Figure 28000-29. Axial Dye Distribution
for a 2-String Fuel Assembly
(Subchannel K)

Figure 28000-30. Axial Dye Distribution for a
2-String Fuel Assembly
(Subchannel A)

flow to axial flow, determined by salt injection into multiple channels, is $0.75 \pm 0.25 \text{ ft}^{-1}$. Single channel injection data give an average cross flow ratio of $0.65 \pm 0.16 \text{ ft}^{-1}$, equivalent to a cross flow of $0.44 \pm 0.11 \text{ lb/sec-ft}$ of 120°F water. This compares closely with the cross flow of $0.326 \pm 0.041 \text{ lb/sec-ft}$ previously measured in a dual channel test section.

The larger standard deviation of data shown for the multiple channel tests is attributed to the increased complexity of both the test model and the noise correlation analyses necessary to obtain useful data. It is significant that the data from both tests overlap within the standard deviations. This correlation of data indicates that various flow parameter tests could be conducted in less complex test systems with full bundle model tests being reserved for the final qualification testing program.

F. PHYSICS FUEL PRODUCTION

The objective of this effort is to fabricate aluminum-clad uranium carbide fuel rods for use in the HWO CR experimental physics work to be conducted at the Savannah River critical experiment facility. Approximately 2000 kg of 1%

enriched hyperstoichiometric UC will be cast and clad with 6061 aluminum alloy. A total of 1155 rods, each with a fuel column height of $48 \pm 1/8$ in. will be fabricated.

1. Fuel Materials

All raw materials were received and approved for use. The product drawings and specification were approved and released, and all necessary approvals for start of production were obtained including the nuclear safety analysis.

2. Fuel Fabrication

Production of uranium carbide feed material was initiated on December 20, 1966 and the production rate has progressed at the scheduled rate of one heat per working day. This process involves blending the UO_2 and graphite powder and reacting the agglomerate at high temperature in a vacuum induction furnace. Seventeen heats of feed stock material are required to produce the fuel required. Acceptance analyses of the finished product is delayed due to failure in the equipment required for the analyses of oxygen and nitrogen.

3. Cladding

All the fuel rod cladding material was delivered and passed inspection, including eddy current testing. Plans were initiated to have the cladding cut to length and chemically cleaned in preparation for fuel rod fabrication.

4. End Caps

The rod end caps which require an anodized coating on a portion of the exposed surfaces were ordered. An initial shipment of caps was received and rejected due to improper location of the anodizing.

5. Fuel Rod Assembly

The original plan for end closure welds was based on TIG welding to be performed in a retort containing a mixture of helium and argon gas. Welding development work initiated during the quarter revealed that leak tight welds could not be obtained with the proposed process and equipment. The cladding material, 6061-T6 aluminum alloy is very susceptible to cracking during welding, and this problem is aggravated by the weld geometry that required a relatively thin wall tube to be welded to a heavy section end cap. A variety of other welding

processes and equipment variations were experimentally evaluated to determine a method of making the welds. Based on the results of these experiments, electron beam welding was selected as the method for fabricating both end closures on the fuel rods. No problems have been encountered in achieving leak tight welds, that show no cracking upon dye penetrant inspection, using the electron beam process. All TIG welding processes evaluated showed severe problems with surface cracking and leak tightness. Process development to determine the optimum electron beam welding parameters and to evaluate the use of weld filler material is in process. Detailed metallographic examinations of sections of the developmental electron beam welds show a tendency toward microcracks at the root of the weld, and the weld development is aimed at eliminating or minimizing this condition.

III. NEXT REPORT PERIOD ACTIVITIES

A. SAP DEVELOPMENT

Tests for characterization of billet material now on hand will be completed. Tests on final shipments of billet material will be initiated. Evaluation of characteristics of various lots of SAP billet stock received for the HWOCR development program will be initiated.

The trial extrusion of finned SAP 895 cladding with spiralled fins will be completed. Evaluation of this cladding will be initiated. Statistical analysis of data from the current tube burst stress-rupture tests will be completed.

The test procedure for cladding tube nondestructive test development will be published.

B. FUEL ASSEMBLY FABRICATION DEVELOPMENT

Initial weld parameters will be established for the U-307 MFW end closures. Preliminary trials will be made using SAP 895 cladding of the EXP-WR1-953 geometry. When the U-307 design is fixed, cladding will be procured to that geometry for further MFW trials.

A series of MFW closures will be made to evaluate the effect of synchronizing peak interface temperature and forging force on closure strength.

Data will be obtained on the high resolution ultrasonic inspection of MFW closures which have the cladding fins extending up to the weld joint.

The cold-heading work for attaching the end caps to the end plates will continue. Tensile tests will be performed to determine the strength of the mechanical joint between the end caps and end plate. Design of the grooves in the countersunk surface of the end plate will be altered and more test specimens will be fabricated to determine if rotation between the fuel rods and end plate can be prevented.

C. IRRADIATION TESTING

1. NRU Irradiations

a. U-305 Post-Irradiation Examination

The examination of bundle LTA will be completed. Metallography of cladding, fuel and composite fuel/cladding wafers will be conducted to further evaluate fuel rod performance and to study the fuel/cladding interaction.

b. EXP-NRU-305-A1 Operation

U-305-A1 coolant chemistry tests will be continued. Low water tests which were not concluded during the last report period will be completed during the next period.

2. WR1 Irradiations — EXP-WR1-951 Operation

The operation of EXP-WR1-951 will continue. It is planned to return the bulk coolant temperature to its original level (335°C inlet). Analyses of the film formation behavior will be continued.

3. Design and Fabrication of WR1 Test Fuel Elements

a. EXP-WR1-952

Bellybands will be designed and installed on the 952 string in preparation for insertion.

b. EXP-WR1-954

Fabrication of the 954 string, including bellybands, will be completed. Documentation of the design, analysis and pre-irradiation data will be initiated. The test procedure will be completed.

c. EXP-WR1-953

Existing records, drawing, and pre-irradiation data for the 953 will be stored for possible future use.

4. Design and Fabrication of NRU Test Fuel Elements -- EXP-NRU-307

Pending resolution of design uncertainties regarding cladding sizing and bellybanding configurations, the final design of the U-307 will be initiated. A cladding order will be placed and fuel fabrication initiated.

D. OUT-OF-PILE TESTING

1. SAP-UC Compatibility

The current SAP-UC compatibility test series will be continued. An additional capsule test series will be designed to study the effect on SAP-UC compatibility of (a) minimizing the oxide film on the SAP component, and (b) embedding chips of UC in the surface of the SAP component.

2. Hydraulic Testing -- Coolant Mixing Tests

Additional tests will be run to determine the effect of fuel bundle loading on mixing in the peripheral channels. All the data will be reduced and analyzed for inclusion in a Technical Information Report.

All other coolant mixing tests have been discontinued and loop operations are suspended indefinitely.

E. PHYSICS FUEL PRODUCTIONS

The following activities are scheduled for the next period.

- 1) Fuel rod end closure weld process will be qualified.
- 2) Casting and machining of UC fuel slugs will be completed.
- 3) Fuel rod assembly product will be completed.
- 4) Completed fuel rods will be shipped to SRL.

1

2

3

4

5

HWOCR QUARTERLY REPORT

TITLE: Large Component Testing

WBS NO: 31000

REPORTING PERIOD: October through December 1966

Work on WBS No. 31000 was suspended for the balance of the fiscal year.

1

2

3

4

5

HWO CR QUARTERLY REPORT

TITLE: Organic Coolant Evaluation and
Analytical Techniques

WBS NO: 35110

REPORTING PERIOD: October through December 1966

I. PROJECT OBJECTIVES

The objective of this activity is to provide information on the properties and chemistry of organic coolants as required by the HWO CR program. This will be accomplished by evaluating available data and by new experimental work where required. Specific areas to be covered are:

- a) Physical properties
- b) Decomposition and makeup rates
- c) Chemical analysis and testing
- d) Coolant oxidation
- e) Halogen chemistry

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. PHYSICAL PROPERTIES

Effort in this area centered on the preparation of a topical report covering completed studies on coolant melting points, transition energies and specific heat measurements using differential scanning calorimetry. The report will be completed during the next report period.

B. DECOMPOSITION AND MAKEUP RATES — MIT IN-PILE LOOP FACILITY

Since June 1966, the MIT loop has been undergoing modifications to the main loop system in preparation for the high fast neutron runs. Modifications include installation of a differential pressure cell for surge tank level monitoring while the sampling and makeup pumps are operating, installation of lines from the primary system to the pumps, and removal of an organic cooler.

The irradiation of Santowax WR coolant at low fast neutron fraction ($f_n = 0.071$) was summarized in a topical report.* Decomposition rates, radiolysis reaction order, the magnitude of the fast neutron effect, and radiopyrolysis are discussed in the report.

Irradiation of Santowax OM coolant at $f_n = 0.40$ and 84% terphenyl ($\sim 10\%$ HB) at temperatures of 572, 700, 750, and 800°F was to constitute the next series of tests. These four tests were to be followed by tests on Santowax WR coolant under the same conditions so that a direct comparison of decomposition rates could be made under conditions approximating HWOOCR conditions. However, a new sampling and makeup system was required because of the higher dose rate in the MITR position No. 1 (higher dose rate partly the result of increase of the reactor power level from 2 to 5 Mw); due to the problems encountered in achieving satisfactory operation of metering pumps in the system, the only irradiations presently feasible are those in which the cycle time for sample removal, distillation, and return to the loop system is at least 16 hr. The originally planned run 19 (Santowax OM coolant, 572°F, transient to 84% terphenyl) was therefore modified to 63% terphenyl to obtain irradiation results while work is in progress on the sampling and makeup system.

This run, designated 19a, was initiated on December 2. The HB content is 27%, the average dose rate to the organic in the capsule is 1.47 w/gm, and the fast neutron fraction is 0.35. Steady-state conditions are maintained by distilling 300-gm samples of irradiated coolant and by combining the distillate with Santowax OM makeup to return to the loop. An HB cutoff distillation is employed. Run 19a is expected to give additional information on the reaction order of radiolysis. The next steady-state run (No. 20) is expected to be an irradiation of Santowax OM at 572°F, 84% terphenyl ($\sim 10\%$ HB) under the same conditions of test neutron and dose rate as run 19a.

C. COOLANT OXIDATION

Operation of the oxidation loop was initiated by conducting a series of thermal oxidation runs without irradiation and using freshly distilled, unirradiated Santowax OM coolant. Several hardware difficulties were encountered and corrected

*"Effect of Reactor Irradiation on Santowax WR," MIT-334-70, October 1966

in these first runs. Typical difficulties included (1) inadequate temperature sensing and control, (2) inadequate gas flow and pressure control, (3) massive mass transfer of warm coolant via vaporization and condensation in both the inlet and outlet lines.

The loop was operated at a maximum temperature of 600°F, and higher temperature runs will be attempted in the near future. The first series of thermal oxidation runs was conducted at 300°F. The steady-state oxygen up-take was measured over a period of 19 hr for one charge of distilled Santowax OM coolant, but the test was terminated as the result of a malfunction of the coolant pump. The pump failed because a piece of slag from the original casting was left within the remote diaphragm head during pump fabrication. Damage to the stainless steel diaphragm resulted in severe coolant leakage. Operation of the loop has been suspended pending receipt of replacement parts.

A gas chromatograph time-of-flight mass spectrometer train has been assembled for use in identifying the main coolant oxidation products.

A heater for the Van de Graaff irradiation capsule is under construction. The capsules and capsule loader have already been completed.

D. HALOGEN CHEMISTRY

In connection with a literature survey concerning the effect of chlorine on film formation in organic coolants, a review was made of the information available on the presence of chlorine in the Piqua reactor coolant during the reactor's first 2-1/2 yr of operation. It is hoped that a determination of the concentration of chlorine in the coolant during PNPf operation will lead to a better understanding of the maximum acceptable concentration with respect to film formation effects. Six samples of PNPf circulating reactor coolant were selected as representative of the various periods during the 2-1/2 yr of operation. These have been analyzed in triplicate for chlorine by the recently developed neutron activation, gamma-gamma coincidence counting technique. Results are tabulated on the following page. From the tabulation, the range of values for chlorine in PNPf coolant appears to have been between 0.078 to 0.29 ppm. It is of interest to compare these results with the chlorine concentration found in the PNPf outer moderator deposits.

<u>Sample Date</u>	<u>Chlorine by Coincidence Counting (ppm)</u>
3-17-64	0.19
4-6-64	0.29
10-30-64	0.078
12-22-64	0.093
6-16-65	0.079
11-22-65	0.178

Analyses of a number of outer moderator deposits for their iron and chlorine contents were carried out and reported earlier. Iron content was in the range from 300 to 1100 ppm, a typical value being 700 ppm. Chlorine content range was from 10 to 50 ppm with a typical value of 30 ppm. No values are currently available on the chlorine content of fuel element deposits. Comparison of these values to each other, and to the values in the circulating coolant, (to establish if iron and chlorine are concentrated to the same extent in the moderator deposits) is given below.

	<u>Coolant</u>	<u>Deposit</u>	<u>Concentration Ratio Deposit/Coolant</u>
Chlorine (ppm)	0.078 - 0.29 (avg 0.15)	avg = 30	200
Iron (ppm)	2 - 3 (avg 2.5)	avg = 700	280
Atom ratio, Fe/Cl	11	17	

These data show that the chlorine and iron are concentrated in the deposit to about the same extent and that the atom ratio of Fe to Cl is about the same in the coolant and deposit. These observations are compatible with conclusions of the Canadian workers about the role of a thermally unstable chlorine-iron-organic complex in the film deposition mechanism. According to their conclusions, this complex was a means of abstracting iron from the piping walls, transporting it through the coolant, depositing the complex onto the heat transfer surfaces, and releasing the chlorine back to the coolant as the result of thermal decomposition of the complex. The Canadians also concluded that, when the chlorine content of the coolant was maintained below 3 ppm, the fuel element deposition rate was acceptably low.

Based upon the data discussed above, the role of both iron and chlorine (or an iron-chlorine-organic complex) does not appear to have been a major factor, in outer moderator deposit formation, because of the low concentrations of iron in the deposits.

III. NEXT REPORT PERIOD ACTIVITIES

A topical report will be prepared summarizing the measurements of melting points and specific heat of coolants using the differential scanning calorimeter.

Recommendations will be formulated on an experimental program to resolve operational questions on chlorine effects during reactor operation.

The coolant oxidation rate will be measured on Santowax OM as a function of temperature and radiation dose in the cobalt-60 loop.

.

.

.

.

.

HWO CR QUARTERLY REPORT

TITLE: Parameters Affecting Film Formation

WBS NO: 35120

REPORTING PERIOD: October through December 1966

I. PROJECT OBJECTIVES

The objectives of this activity are to develop a detailed, quantitative understanding of the film formation processes which take place in reactors of the HWO CR design; explore possible methods of reducing film formation rates; and complete the design and test of a film formation rate meter intended to measure "in situ" the film forming tendency of organic coolants. A forced circulation loop (film formation loop) will be operated to evaluate the effect on film formation rates of such parameters as radiation type and intensity, coolant type and composition (including impurities), and coolant flow rate. By developing an understanding of the film formation micro-processes, it will be possible by direct engineering techniques to evaluate the macro-processes taking place in the nuclear reactor. In-pile tests using a fuel element will be conducted in the U-3 loop. A thermal coking test is in process to determine the effect on irradiated coolant of long term (up to 1 yr) exposure to elevated temperatures. The evaluation of the film formation data from OMRE, PNPf, U-3, WR-1, and MIT loop will continue.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. FILM FORMATION LOOP

During the report period, effort was concentrated on the items listed below.

- 1) Completion of the design package for the heat rejection system (assembly is approximately 50% complete).

- 2) Completion of the design of the control console (assembly is approximately 80% complete).

3) Completion of the stress and thermal analysis of the coaxial heater section. Several design changes were indicated.

4) Release of engineering orders for the assembly of the loop flow meter, filter, sample containers, and heater section electrode hardware.

5) Initiation of construction drawings for the assembly of the loop.

6) Completion of tests on the flow control valves. Analysis of the results indicated that no problems are anticipated in flow regulation.

7) Inspection of the pressure transmitters in the pressure differential systems. Corrosion was observed in two of the three transmitters which were dismantled. The vendor indicated that, contrary to the requirements stated in the purchasing specification, the units were degreased with carbon tetrachloride prior to acceptance testing. Apparently, during the high temperature phase of acceptance testing, the residual carbon tetrachloride attacked the stainless steel components. The vendor has indicated that the damaged parts will be replaced.

8) Completion of the low temperature (500°F) phase of the acceptance tests on the temperature differential system. Final acceptance is pending the results of the high temperature (800°F) tests.

9) Completion and incorporation of an analytical test procedure for chloride ion contamination into the preassembly cleaning specification for the loop.

Further delays in the delivery of the main circulating pump have developed. The vendor indicates that shipment of the pump should be made by the end of January 1967. However, no programmatic delays should occur, as the limiting item is the approval of the proposed contract for the irradiation site.

B. THERMAL COKING TESTS

Thermal coking tests were continued during the report period. The 3-month test duration capsules were removed from test. The gas within the capsules was sampled, but has not as yet been analyzed.

C. U-3 LOOP FILM FORMATION TESTS

Due to unstable reactor operation and a 6-week maintenance shutdown, the tests originally planned for this report period were not conducted. A confirming run at low water content (less than 25 ppm) was initiated on December 27.

Three instances of relatively rapid increases in cladding temperatures noted during the U-305 and U-305-A1 experiments — July 30 to August 14, 1965 (U-305), September 5 to 8, 1965 (U-305) and August 9 to 19, 1966 (U-305-A1) — were analyzed. Each corresponded to operating periods in which the water

content of the coolant was low (less than 50 ppm). An analysis of the temperature data for the different thermocouples shows the best correlation between film formation and temperature is obtained by comparing absolute temperature rise, rather than film formation parameters, with the initial (clean surface) temperature. Temperature data from thermocouples on similarly located fuel rods followed an approximate Arrhenius relationship ($\log \Delta T = A - B/T$) when compared in this manner. All the temperature data were brought into rough agreement when the temperature rise was normalized by dividing by the square of the relative heat generation rate for each rod. This procedure assumed that the film thickness was proportional to the radiation dose, and that the radiation dose was proportional to heat generation.

It is noteworthy that the data from the two U-305 episodes fit the same Arrhenius line even though the surface temperatures varied as much as 25°C. This would indicate that the film formed in the first episode may have changed so that film formation was again occurring at a surface whose temperature was the same as the initial cladding temperature.

The Arrhenius plots for the U-305 and U-305-Al data were quite different both in intercept and slope, suggesting a difference in the chemical processes involved. The U-305-Al film formation rate is a factor of 20 lower than the U-305 rate at 700°F, indicating a substantial improvement in coolant quality. The activation energy (temperature dependence) for U-305-Al is approximately twice that for U-305.

This analysis assumed a temperature dependence of film formation. Another equally valid analysis could be performed in which no temperature dependence, but rather a higher order dose rate dependence, is assumed. A variety of results could be obtained depending upon the order used. To determine if the temperature effect exists, it is proposed to repeat the low water test and vary the inlet temperature.

D. ANALYSIS OF FILM FORMATION DATA

Because of the lack of understanding of film formation processes, a largely empirical approach, using the model developed for PNPF,* is being used for

*T. T. Shimazaki, T. J. Carter, "Film Formation Model for Organic Coolant Systems," NAA-SR-TDR-12254. November 23, 1966.

the analysis of data from OMRE, PNPf, U-3, WR1, and the MIT Loop. Reasonably good correlation has been obtained among data from PNPf. Measured film thicknesses from OMRE, U-3, and the MIT Loop are lower by factors of 3 to 20 than those predicted by the model. This may indicate that the model is not accurate for conditions very different from PNPf, or that the film forming species are different (for example in particle size).

III. NEXT REPORT PERIOD ACTIVITIES

A. FILM FORMATION LOOP

Assembly of the Film Formation Loop will be initiated.

B. THERMAL COKING TESTS

Analysis of the three-month capsules will be completed. Six-month capsules will be removed from test, and analyses will begin.

C. U-3 FILM FORMATION TESTS

A low water content test will be performed to confirm the water effect previously noted. This test will be followed by a test at very high water (~800 ppm) to determine possibility of removing a film in this manner.

D. EVALUATION OF FOULING DATA

Effort will be continued on the interpretation of film formation results.

HWO CR QUARTERLY REPORT

TITLE: Coolant Technology Loop

WBS NO: 35700

REPORTING PERIOD: October through December 1966

I. PROJECT OBJECTIVES

The objective of this activity is to design, construct and operate a Coolant Technology Loop (CTL) for in-pile investigation of organic coolant behavior. The CTL will be the only in-pile loop in the HWO CR program specifically designed for organic coolant technology tests, without the restrictions imposed by fuel element tests. It will combine in one facility several proposed in-pile irradiation activities, with advantages in economy and efficiency. These activities include in-pile confirmation of the Film Formation Loop parametric studies, oxygen and chlorine effects on film formation, study of film flaking, purification studies including new methods, tests of coolant quality instrumentation and the oxidation and film forming properties of hydrocracked coolant.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. LOOP DESIGN

Title I activity was authorized and preliminary planning was conducted. The following design features, if they prove to be technically feasible, are proposed as highly desirable for the CTL:

- 1) Two or more parallel in-pile sections, so that different values of temperature, radiation, and flow parameters can be compared simultaneously, with identical coolant and identical reactor operating history.
- 2) Precise instrumentation and control, including reliable (and preferably replaceable) temperature sensors to measure surface and coolant temperatures without perturbing coolant flow, and pressure sensors to measure fuel pressure drop.
- 3) Variable dose rates, including fast neutron dose rates up to about 3 w/gm (bracketing HWO CR peak value of 2.4 w/gm)

4) Fast neutron flux supplied largely by a separately cooled annular fuel element, so that a high rate of generation of HB and film forming species can be combined with a relatively small loop heat rejection requirement to minimize out-of-pile volume.

5) Fueled film formation probes, removable for inspection, with an internal heating or cooling system so that temperature can be varied (over a moderate range) independent of radiation flux.

6) A completely tight system with complete control of coolant composition and purity.

7) Modular design for easy addition or replacement of experimental units such as purification system.

B. COOLANT TECHNOLOGY PROGRAM

The coolant technology program was replanned to include the CTL and other proposed activities. The program is primarily oriented to the prediction, measurement, and control of film formation. The information sought is divided into three categories in order of priority: Feasibility information, needed during Title I design of the Demonstration Plant; design information, needed during design and construction; and advanced technology, useful during operation and for future HWO CR plants. The CTL, scheduled for operation in GFY 1969, will contribute to the last two categories.

III. NEXT REPORT PERIOD ACTIVITIES

Title I design will be conducted.

HWOCR QUARTERLY REPORT

TITLE: Coolant Purification and Recovery

WBS NO: 67000

REPORTING PERIOD: October through December 1966

Work on WBS No. 67000 was suspended for the balance of the fiscal year.

1

2

3

4

5

HWOCR QUARTERLY REPORT

TITLE: On-Power Refueling

WBS NO: 81000

REPORTING PERIOD: October through December 1966

I. PROJECT OBJECTIVES

The overall objective of this task is the design and evaluation testing of the on-power refueling machine and its auxiliary systems under all conditions of operating environment except radiation. Testing will be performed in a logical, step-by-step manner with initial emphasis on (a) concept evaluation testing in critical interface areas between the refueling machine and reactor pressure tubes, (b) fuel and other components to be manipulated, and (c) auxiliary fuel handling systems. The results of these tests will form a basis for the prototype design or selection of major machine components. Selected components will then be fabricated and design proof-tested under environmental conditions. The final phase of the development program will be the prototype testing of the assembled refueling machine under operating conditions.

II. PROGRESS DURING REPORT PERIOD AND EVALUATION
OF WORK TO DATE

A. REFUELING MACHINE DESIGN

The refueling concept referencing latching shield plugs as the means of fuel retention within the coolant channel was further developed with primary emphasis on refueling head components. A new End Item Specification for the refueling machine was drafted and sub-system concepts developed for a tool post hydraulic drive and control of the machine environment.

1. Refueling Head Component Conceptsa. Tool Post

Specific force and motion sequences required for ram actuation of the fuel channel end fitting and latching shield plug were developed and geometries were selected to be compatible with both component actuating mechanisms and

ram-end latching mechanisms. Latching fingers were selected for the latter in preference to ball detents in order to reduce contact stresses and brinelling.

Vendor recommendations were received for ball screw and ball spline units to drive the three tool post rams, and materials were suggested for prototype application. Although producibility of these units in the required length appears feasible, material selection will be deferred pending test results of a short stroke test unit operating in organic coolant.

b. Magazine

A 12-chamber rotating storage drum was selected for the storage and transport of fuel and pressure tube components. Chambers have been assigned as listed below.

- 1) Fuel, 5
- 2) Fuel channel end closures (one containing a spare closure), 2
- 3) Latching shield plugs (one containing a spare upflow plug; one, a downflow), 3
- 4) Inner shield plugs (one containing a spare plug), 2

Chambers are equipped with latches for the retention of fuel and channel components during storage residence.

A gating mechanism, suitable for use with flat end plate fuel, has been incorporated into the magazine assembly. This device holds the fuel column while fuel or channel components are being removed from, or added to, the column. Two additional functions are provided by the gating subassembly:

- 1) Mechanical separator for pushing fuel into engagement with storage chamber latches.
- 2) Mechanical sensor for indicating the presence of two abutting fuel elements as a prerequisite to engagement of the fuel gate.

Ultrasonic sensing techniques are also being explored as an alternate to mechanical sensing.

c. Snout Coupling

The availability of suitable commercial components for this assembly has been explored with particular reference to the valve providing the machine closure function and to the radial clamp providing the machine-to-pressure tube coupling function. Components for both functions are available to meet anticipated requirements.

A metallic, self-energizing seal has been selected for prototype machine-to-pressure tube sealing. Factors considered in this selection are listed below.

- 1) Availability of relatively high axial seating loads for forcing through any organic residue on seal mating surfaces
- 2) Resistance to damage due to thermal cycling
- 3) Simplicity and ruggedness
- 4) Reduced alignment requirements
- 5) Make/break duty cycle experience in other applications.

2. Refueling Machine Subsystems Concepts

a. Tool Post Drive

A hydraulic drive system has been selected for operation of the tool post rams. During column movement, the rams will operate at a nominal speed of 10 fpm and will have varying force requirements. Each machine, upper and lower, must be capable of providing two sets of forces, one for an upflow channel, and one for a downflow channel. During the course of column movement, the column grows or retracts as a function of temperature. This variation is expected to amount to 0.250 in. over the full stroke of ~ 30 ft. An additional operating requirement of the ram is to maintain force at zero speed (or stall condition). This condition occurs frequently; during the first 6 min of the refueling cycle the ram drive is stalled for about 4 min, exerting maximum force.

Two basic systems were studied to meet these requirements: a torque drive and a synchronized drive. The torque drive could be either electrical or hydraulic, the synchronized drive would be electrical.

In the torque drive, one ram is speed-controlled and the opposing ram is torque or force-controlled. As the speed control ram is moved, the force-controlled ram follows to maintain the prescribed force. The following action of the latter ram as a function of force on the column automatically adjusts the system to column length variation.

With the synchronized drive, the rams are locked together by an external circuit and, when set in motion, are controlled by speed and position demand. A resilience (or spring) in series with the rams and column maintains the prescribed force as the length of the fuel column varies.

AI-CE-56
WBS 81000

TABLE 81000-1
SNOUT SEAL TEST RESULTS

	Harrison-K (TS 50-5-01)	Haskel-K (TS 50-5-04)	Dover-W (TS 50-5-05)	Pressure Science-E (TS 50-5-06)	Parker-V (TS 50-5-10)
<u>400° F Run</u>					
Time (hr)	25	25	25	25	25
Average temperature (°F)	410	409	400	401	401
Average pressure (psig)	294	293	300	295	295
Average flow (gpm)	5.2	5.3	6.6	6.1	6.1
Leakage (cc/hr)	0.0	0.0	0.0	0.0	0.0
<u>750° F Run</u>					
Time (hr)	24	24	26	25	25
Average temperature (°F)	757	758	731	752	754
Average pressure (psig)	300	300	300	300	300
Average flow (gpm)	5.5	5.5	5.4	7.8	7.8
Leakage (cc/hr)	0.0	0.0	0.0	0.0	0.0
<u>Thermal Cycling</u>					
Number of cycles	25	25	25	25	25
Average maximum temperature (°F)	698	690	707	690	694
Average minimum temperature (°F)	405	405	415	407	406
Average pressure	300	300	299	298	298
Average leakage (cc)	2.0*	0.0	0.0	0.0	0.0

*Total leakage occurring during the 14th and 15th thermal cycles.

TABLE 81000-2
STATIC FACE SEAL DATA

Seal	Deflection (in.)	I.D. (in.)	Load (lb)	Flange Finish Top-Bottom (rms)	Set (in.)	Material
Harrison-K (TS 50-5-01)	0.0165	4.907	2,460	15-14	0.0132	Inconel-X Gold Plate
Haskel-K (TS 50-5-04)	0.0073	4.898	1,410	12-20	0.0028	Inconel-X Gold Plate
Dover-W (TS 50-5-05)	0.0068	5.250	5,520	8-8	0.032	17-4PH SS Gold Plate
Pressure Science-E (TS 50-5-06)	0.0143	4.859	870	15-14	0.0098	Inconel-X Silver Plate
Parker-V (TS 50-5-10)	0.0114	5.125	19,050	12-20	0.0099	Inconel-X Silver Plate

In both schemes, force limit overrides on speed controls and ram position comparison provide checks on the system to assure safe operation.

In general, the study shows the synchronized drive offers no significant advantage. However, certain advantages are evident with the torque drive. These advantages are primarily the flexibility of the system to accommodate the changing of operating forces, the simpler control mode where only one drive is speed-controlled, and the applicability of this type of drive to other drives on the refueling machine. The selected hydraulic drive has the additional significant advantage of utilizing components that are essentially off-the-shelf, are compact in size, and are readily operated in the environment.

B. FUEL TRANSFER EQUIPMENT DESIGN

Conceptual designs were made for the details of the fuel transfer shuttle system and of the grapple and hoist used to lower the spent fuel from the shuttle system into the storage tube. Since the flat end plate fuel assembly has minimal provision for grappling, a detailed study was made of the grapple, and preparation was begun of drawings for fabrication of a model to verify its operation.

C. FUEL HANDLING MACHINE DEVELOPMENT AND TESTING

1. Snout Seal Testing

The testing of the first generation of snout-pressure tube seal candidates has been completed. Five types of static seals were tested and their performance was satisfactory. The testing, performed in TF-50 in Santowax OM, consisted of the tests listed below.

- 1) 25 hr run at 400°F
- 2) 25 hr run at 750°F
- 3) 25 thermal cycles from 400 to 750 to 400°F at a nominal temperature ramp of 6°F/second.

A tabulation of test results is shown in Tables 81000-1 and 81000-2. Table 81000-1 lists the test parameters and leakage rates while Table 81000-2 contains the deflection, loads and permanent set on unloading of the five seals tested.

Additional planned phases of the snout seal test program have been deferred until the next fiscal year.

TABLE 81000-3
ORGANIC SOAK PERFORMANCE
(In Santowax OM at 400°F)

	Length of Time (hr)	Force to Begin Actuation Normalized to Force Required Clean Cold in Air (1.0 = 121 lb)	Force to Fully Actuate Normalized to Force Required Clean Cold in Air (1.0 = 246 lb)
PHASE I	0	0.76	0.61
	13	0.88	0.65
	120	0.88	0.74
PHASE II	0	0.43	0.57
	24	0.35	0.84
	25	0.37	0.88

NOTE: Mechanism was not cleaned between Phase I and Phase II.

TABLE 81000-4
ORGANIC BAKE PERFORMANCE
(In Air at 400°F)

	Length of Time (hr)	Force to Begin Actuation Normalized to Clean Cold in Air	Force to Fully Actuate Normalized to Clean Cold in Air
PHASE III	0	0.34	0.80
	200	1.55	0.61
PHASE IV	0 hr 400°F air (after 700-hr OM boil-off)	0.89	1.11
	200	3.00	0.93

NOTE: Mechanism was not cleaned between Phases II and III but was cleaned between Phases III and IV.

2. CANDU End Seal Mechanism Coking Tests

The CANDU End Seal Mechanism coking tests were performed to provide qualitative information of the effect on actuation force by:

- 1) Soaking in 400°F Santowax OM,
- 2) Baking at 400°F in air with a Santowax film, and
- 3) Boiling off Santowax at 400°F followed by baking in air at 400°F.

Tables 81000-3 and 81000-4 summarize the test results. Most of the normalized forces are less than 1 because elevated temperature and the presence of organic lower the activation forces over the clean cold condition. Figure 81000-1 shows several of the "coked" parts of the Mechanism after Phase IV of the test program. In general, none of the aforementioned tests resulted in a wide variation in the force required to fully actuate the mechanism. The significant results of the program were:

- 1) Breakaway force increases significantly with baking time in air boil-off of organic rather than by draining.
- 2) When the mechanism was cooled to room temperature after Phases III and IV, it could not be actuated (maximum force tried was 4 times the normal force required).
- 3) The cold frozen mechanism could be freed by heating it to approximately 200°F.

These results indicate that a test program is required to study whether or not a similar organic coking problem exists for the HWO CR end seal mechanism.

3. Ball Screw Testing

Performance and design requirements for a prototype ball screw and ball spline unit for the refueling machine tool post were prepared and vendor quotations were requested. A short ball screw unit for wear testing in hot organic was included as part of this quotation package, and test requirements were established.

All major components for the ball screw test unit, except the ball screw itself, are ordered. The organic system design and the test housing support structure were completed. The entire assembly was designed using the modular

Figure 81000-1 CANDU Mechanism
Sliding Surface After Baking
in Santowax-OM

approach for easy portability and maximum flexibility. The four basic modules are:

- 1) Test Housing
- 2) Organic System
- 3) Control System
- 4) Hydraulic System

4. Mechanical Couplings, Ball Valves, Flexible Hose, and Ball Bearing Testing

These tests were postponed until the next fiscal year when they will again be reviewed. Status reports were written and issued on the ball valve and flexible hose tests and they are being prepared for the mechanical coupling and bearing tests.

III. NEXT REPORT PERIOD ACTIVITIES

A. REFUELING MACHINE DESIGN

Critical component specifications for the performance, design, test, and qualification of the tool post, magazine, snout coupling, environmental control system, refueling head suspension system, and the control and instrumentation system will be prepared. Prototype design layouts will be initiated for components of the refueling head and procurement action initiated for prototype ball screw and bearing test units.

B. REFUELING MACHINE DEVELOPMENT

The following development activities are planned for the next report period:

- 1) A report will be issued on the results of the Snout Seal Tests.
- 2) A report will be written and issued on the results of the CANDU Mechanism Coking Tests.
- 3) Erection of the ball screw test facility will be initiated.
- 4) Status reports will be issued for the Mechanical Coupling and Ball Bearing Test Programs.

1

2

3

4

DISTRIBUTION LIST
FOR
HWO CR QUARTERLY PROGRESS REPORT
OCTOBER THROUGH DECEMBER, 1966
AI-CE-56

	No. Copies
Advanced Planning & Research Southern Services, Inc. 600 North 18th Street Birmingham, Alabama 35203 Attention: Mr. Ruble A. Thomas, Manager	1
Atomic Energy of Canada, Limited: USAEC Technical Representative Whiteshell Branch, Chalk River Office U. S. Atomic Energy Commission c/o Whiteshell Nuclear Research Establishment AECL Pinawa, Manitoba, Canada Attention: Mr. H. J. Reynolds	1
USAEC Scientific Representative Chalk River Liaison Office c/o Atomic Energy of Canada Limited Chalk River, Ontario, Canada Attention: Mr. Miller N. Hudson, Jr.	3
Whiteshell Nuclear Research Establishment c/o Atomic Energy of Canada, Limited Pinawa, Manitoba, Canada Attention: Dr. R. F. S. Robertson	1
The Babcock and Wilcox Company Atomic Energy Division P. O. Box 1260 Lynchburg, Virginia 24505 Attention: Mr. W. M. Vannoy	3
Carolinas Virginia Nuclear Power Associates, Inc. Parr, South Carolina 29066 Attention: Mr. H. T. Babb, Manager	1

Communaute Europeenne
De L'Energie Atomique
Ispra Research Center
Casella Postale 1
Ispra (Varese) Italy

Attention: Mr. Tauch

1

Duke Power Company
P. O. Box 2178
Charlotte, North Carolina 28201

Attention: Mr. E. C. Fiss

1

E. I. du Pont de Nemours and Company
Explosives Department
Wilmington 98, Delaware 19898

Attention: Dr. D. F. Babcock, Director
Reactor Engineering Section

1

E. I. du Pont de Nemours and Company
Savannah River Laboratory
Aiken, South Carolina 29801

Attention: Dr. J. L. Crandall
" Mr. S. W. O'Rear, Supv.
Technical Information Serv.

1

2

Massachusetts Institute of Technology
77 Massachusetts Avenue
Room 24-109
Cambridge 39, Massachusetts 02139

Attention: Dr. Manson Benedict, Head
Dept. of Nuclear Engineering

1

Oak Ridge National Laboratory
Union Carbide Corporation
AEC Operations
P. O. Box X
Oak Ridge, Tennessee 37831

Attention: Mr. D. A. Douglas
" Mr. D. E. Ferguson
" Mr. J. A. Lane (ORNL P. O. Box Y)
" Dr. I. Spiewak

1

1

1

1

Pacific Northwest Laboratory
P. O. Box 999
Richland, Washington 99352

Attention: Dr. J. J. Cadwell, Manager
Metallurgy Department

1

Puerto Rico Water Resources Authority
Box 4267
San Juan, Puerto Rico

Attention: Dr. Modesto Iriarte, Jr. 1
Sr. Electrical & Nuclear
Power Engineer

Rensselaer Polytechnic Institute
Troy, New York

Attention: Dr. P. B. Daitch 1
Dept. of Nuclear Science
and Engineering

Tennessee Valley Authority
303 Power Building
Chattanooga, Tennessee 37401

Attention: Mr. C. H. Waugaman, Chief 1
Power Research Staff
" Mr. Harold L. Falkenberry 1

U. S. Atomic Energy Commission:

Brussels Office
U. S. Mission to the European Communities
APO, New York, New York 09667

Attention: Theodore J. Iltis, 1
Acting Sr. USAEC Representative

California Patent Group
P. O. Box 808
Livermore, California 94551

Attention: Mr. F. A. Robertson, Chief 1

Chicago Operations Office
9800 South Cass Avenue
Argonne, Illinois 60439

Attention: Mr. D. E. Simpson, Director 2

Division of Compliance, Region 111
Suite 410 Oakbrook Professional Building
Oak Brook, Illinois 60523

Attention: Mr. H. D. Thornburg 1

Oak Ridge Operations Office
P. O. Box E
Oak Ridge, Tennessee 37830

Attention: Mr. H. W. Behrman 1
" Mr. R. L. Philippone 1

U. S. Atomic Energy Commission (continued):

Richland Operations Office

P. O. Box 550

Richland, Washington 99352

Attention: Mr. P. G. Holsted, Director
 Research & Development Div.

2

San Francisco Operations Office

2111 Bancroft Way

Berkeley, California 94704

Attention: Mr. Ellison C. M. Shute, Manager

2

Headquarters, Washington, D. C. 20545:

Attention: Mr. R. M. Forssell, Chief
 Advanced Development Branch
 Division of Naval Reactors

1

Division of Reactor Development & Technology:

Attention: Mr. Milton Shaw, Director

1

" Mr. R. W. Bean

1

" Mr. M. Booth

1

" Mr. J. W. Crawford

1

" Mr. D. E. Erb

1

" Mr. R. Feit

1

" Mr. A. Giambusso

1

" Mr. P. A. Halpine

1

" Mr. F. Kerze

1

" Mr. J. A. Lieberman

1

" Mr. W. M. McVey

1

" Mr. B. T. Resnick

1

" Mr. M. Rosen

1

" Mr. E. E. Sinclair

1

" Mr. F. R. Standerfer

1

" Mr. A. N. Tardiff

2

" Mr. K. A. Trickett

1

" Mr. W. W. Ward

1

" Mr. M. J. Whitman

1

" Mr. W. A. Williams

1

" Mr. I. F. Zartman

1

RDT Site Offices:

RDT Program Office (AI-CE)

CPAO

Canoga Park, California

Attention: Mr. R. L. Morgan
 RDT Sr. Site Representative

4

RDT Site Offices (Continued):

RDT Site Office
P. O. Box 500
Windsor, Connecticut

Attention: Mr. A. J. Alexander
RDT Sr. Site Representative

3

RDT Site Office
P. O. Box A
Aiken, South Carolina

Attention: Mr. J. H. Kruth
RDT Site Representative

1

United States Department of the Interior
Bonneville Power Administration
P. O. Box 3621
Portland, Oregon 97208

Attention: Mr. E. C. Starr
Consulting Engineer

1

Washington Public Power Supply System
P. O. Box 166
130 Vista Way
Kennewick, Washington

Attention: Mr. Owen W. Hurd,
Managing Director

1

Westinghouse Electric Corporation
Atomic Power Department
P. O. Box 355
Pittsburgh 30, Pennsylvania 15230

Attention: Mr. S. Bartnoff,
CVTR Project Manager

1

Mr. Hood Worthington
Devon Apartments, No. 1512
2401 Pennsylvania Avenue
Wilmington, Delaware 19806

1

Atomics International
P. O. Box 309
Canoga Park, California 91304

S. O. Arneson	D/782	C001	1
R. Balent	D/716	C002	1
A. Bennett	D/799	C002	1
G. D. Calkins	D/745	C001	1
E. M. Chandler	D/782	C001	1
J. C. Cochran	D/784	C001	1
D. J. Cockeram	D/731	C002	1
J. Falcon	D/716	C002	1
J. J. Flaherty	D/797	C002	1
D. C. Fulton	D/710	C002	1
S. Golan	D/793	C002	1
R. B. Gordon	D/782	C002	1
J. Hume	D/717	C002	1
R. T. Keen	D/741	C004	1
G. Koris (Patent Office)	GO	C002	1
J. R. Lewis	D/782	C001	1
J. E. Mahlmeister	D/793	C002	1
E. Mendenhall	D/799	C002	1
W. E. Parkins	D/740	C002	1
T. G. Parker	D/782	C001	1
D. Saxe	D/797	C002	1
S. Siegel	D/793	C002	1
C. Starr	D/797	C002	1
B. J. Thomas*	D/782	Canada	1

*Atomics International Representative at
Chalk River, Canada

Combustion Engineering, Inc.
Nuclear Division
P. O. Box 500
Windsor, Connecticut 06095

C. Andrea	1
F. Bevilacqua	1
W. P. Bigler	1
H. Cahn	1
W. P. Chernock	1
A. W. Cope	1
J. R. Dietrich	1
J. S. Greacen	1
R. S. Harding	1
H. V. Lichtenberger	1
E. H. Luther (Patent Office)	1
D. McLaughlin	1
H. Oslick	1
B. J. Selig	2
S. H. Shippenberg	2
P. Sontaro	1
W. Taylor	1

Continued

Combustion Engineering, Inc. (Continued)

J. C. Tobin	1
R. P. Varnes*	1
S. Visner	1
J. M. West	1
R. H. Young	1
W. H. Zinn	1
CE HWOOCR File (G. Basu)	5
CEND File, W. Simon	1

*HPO Representative at Combustion Engineering
Use CE mailing address

HWOOCR Program Office
c/o Atomics International
P. O. Box 309
Canoga Park, California 91304

V. E. Adler	1
L. E. Anderson	1
M. G. Andrews	1
D. P. Cary*	1
W. S. Flinn	1
D. Graf	1
K. S. Helm	1
B. L. Hoffman	1
D. A. Huber	1
J. Jacobson	1
B. Katz	1
R. W. Keaten†	1
S. Miner	1
W. A. Muller	1
W. B. Murray	1
P. G. Parr	1
R. Quantock	1
M. K. Sanders	1
P. F. Shaw	1
H. I. Starr	1
F. M. Stern	1
C. L. Storrs	1
D. S. Thompson	1
C. A. Trilling	1
R. H. Vollmer	1
H. VonSteiger	1
W. B. Wolfe§	1
HPO Central File	1

*HPO Representative at Pinawa, Canada.

†HPO Representative at Halden, Norway.

§HPO Representative at Douglas Point, Canada.