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HEAVY WATER REACTOR PROGRAM

MONTHLY ACTIVITIES REPORT

DECEMBER 1967

Including Contributions from Programs
Sponsored by the USAEC/AECL Cooperative
Program on Heavy Water Power Reactors

AEC RESEARCH & DEVELOPMENT REPORT

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HEAVY WATER REACTOR PROGRAM MONTHLY ACTIVITIES REPORT

December, 1967

Including Contributions from Programs Sponsored by the
USAEC/AECL Cooperative Program on Heavy Water Power Reactors

AEC Division of Reactor
Development and Technology Programs

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This report is furnished pursuant to the Memorandum of Understanding of June 7, 1960, between the U.S. and Canadian Governments, establishing a cooperative program on the development of heavy water moderated power reactors.

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By
The Staff of the Heavy Water
Reactor Program Office

H. Harty, Manager

January 31, 1968

PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON
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HEAVY WATER PROGRAM MONTHLY ACTIVITIES REPORT

DECEMBER, 1967

Including Contributions from Programs Sponsored by the
USAEC/AECL Cooperative Program on Heavy Water Power Reactors

PREFACE

The Heavy Water Reactor Program Office (HWRPO) monthly report is comprised of information from three sources: Program Office activities, activities sponsored by the HWRP, and activities sponsored under the USAEC/AECL Cooperative Program on Heavy Water Power Reactors. The latter activities, reported in Part III, are reproduced verbatim. In this issue the summary was prepared by HWRP personnel.

PREVIOUS MONTHLY REPORTS

July and August 1967	BNWL-608
September 1967	BNWL-627
October 1967	BNWL-631
November 1967	BNWL-664

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HEAVY WATER REACTOR PROGRAM MONTHLY ACTIVITIES REPORT

DECEMBER 1967

Including Contributions from Programs Sponsored by the
 USAEC/AECL Cooperative Program on Heavy Water Power Reactors

H. Harty

I. SUMMARYA. HEAVY WATER REACTOR PROGRAM
ACTIVITIESProgram Management - 12000

Draft reports of status of HWR technology were completed for several areas of technology. Information presented in these reports is being incorporated in the HWR Program Plan. (12100)

General Studies - 13000

The portion of the status of technology report on the kinetics of D₂O reactors has been completed. It

was determined that the problem areas concerning the kinetics of H₂O reactors and D₂O reactors are very similar and that the technology relative to H₂O reactor dynamics can, for the most part, be readily extended to include D₂O reactors. A number of specific conclusions are presented. (13200)

A series of preliminary cost analyses for pressure vessel type HWR's was made using the Swedish optimization programs BOP and BIOP. The parameters investigated are listed below with a sensitivity analysis for each: (13300)

<u>Parameter</u>	<u>Base Value</u>	<u>Change in Value</u>	<u>Change in Unit Cost of Power</u>
Interest rate	7%	Increase to 12%	+41%
Electric capacity	600 MWe	Increase to 1000 MWe	-10%
D ₂ O price	\$20/lb	Decrease to \$13.50/lb	-5%
Plant factor	0.80	Increase to 0.85	-5%
Fuel enrichment	0.711	Increase to 1.25	-5%
Uranium ore cost	\$8/lb U ₃ O ₈	Increase to \$12/lb	+3%
Reactor pressure	1020 psi	Increase to 1520 psi	No change

The status of technology for HWR fuel cycles was completed. Reviews were made of studies conducted at ORNL, SRL, PNL, AI-CE, B&W, CGE, and AECL. The procedures and techniques used in the foregoing studies, as well as the significant results obtained, were reviewed as a basis for determining further studies for evaluating HWR concepts. A number of recommendations are made for future studies. (13300)

The effect of heavy water costs on power reactor electricity costs were reviewed preliminarily. The benefits that could be realized from reduction of the heavy water inventory and unit cost were considered. (13300)

Research and Development Studies - 20000 thru 99000

The HWRP on pressure tubes has been scoped to qualify Zircaloy pressure tube assemblies as fully tested components for HWR service. Critical areas for primary investigation include flaw growth to failure, mechanical failure criteria, biaxial creep, and strain to failure. Two major results of this work will be the development of Criteria for Pressure Tube Design and Criteria for Pressure Tube Removal from reactor service. (21300)

The survey of the status of heat transfer and fluid flow technology is essentially complete. Present studies in the LWR field should be expanded to include configurations and conditions more directly applicable to HWR's, and past results of light water oriented programs should be re-

evaluated in terms of specific HWR needs. Specifically, studies of loss-of-coolant and emergency core cooling for pressure tube HWR's require further evaluation. (28900)

The status of technology of HWR fuels, both ceramic and metal, is essentially complete. Except for the unique problems associated with the use of rod assemblies in tubed reactors, the ceramic fuel development needs are comparable to those for light water reactor fuels. The unique features include fuel assembly design requirements dictated by pressure tube limitations (size), fuel handling and fuel management needs (fuel length), and thermal-hydraulic considerations (vibration and fretting). A trend toward larger fuel assemblies is indicated and suggests a need for investigating the incentives for using radial enrichment in fuel assemblies. Metallic fuels can be expected to successfully achieve fuel exposures of interest in HWR's. However, verification of the performance capability of prototypic fuels under specific HWR conditions is needed. (28100)

B. USAEC/AECL COOPERATIVE PROGRAM ON HEAVY WATER POWER REACTORS

Heat Transfer and Fluid Dynamics

Improvements and modifications were performed on the computer code COBRA to improve its versatility and convenience. This code was originally developed during FY 67 to analyze local subchannel conditions in rod bundle fuel elements by considering

the effects of turbulent mixing and diversion cross-flow between sub-channels during single and two-phase flow. Sample problems were computed to verify the proper function of the code and the accuracy of the revised mathematical model.

Improvements are also being made in the CROUT5 subroutine used in COBRA to correct functional problems that have been experienced by a few users of this code. These problems have appeared only occasionally when certain types of fuel configurations are considered. (TAC 3.1.6)

During the experimental operation of the 9 ft Uniform Bundle with Simulated End Plates, some data were taken beyond the critical heat flux limit. A graph of the data is presented on page 26. It shows the increase in surface temperature with increased power to be strongly dependent upon mass flow rate. (TAC 3.3.8)

Fuels and Materials Technology

Arrangements are being made for diametral measurements of a Zr-2.5 Nb pressure tube after irradiation to 2×10^{20} nvt (>1 MeV) in PRTR. Four crack propagation test specimens of Zr-2.5 Nb were recharged into PRTR for further neutron exposure. During cyclic internal pressurization of Zr-2.5 Nb tube specimens with multiple flaws, one specimen leaked at half the number of cycles and a much shorter fatigue crack length than two other supposedly similar specimens. Work is underway to determine the cause of this variation in behavior. Acoustic

emission techniques have been developed to the point where they are sensitive detectors of plastic strain in tensile, fatigue, and burst tests. These techniques will now be applied to flaw growth and crack propagation tests on Zr-Nb pressure tubes to focus future development of the techniques and obtain vital data under specific test conditions. (TAC 6.9)

An unirradiated control creep test was conducted on Zr-2.5 Nb at 400 °C and 31.6 kg/mm². The measured creep rate was 1.1×10^{-4} hr⁻¹ and the rupture time was 318 hr. The onset of tertiary creep occurred at approximately 2% strain. (TAC 6.10)

Films on irradiated and unirradiated zirconium alloy specimens were examined by X-ray diffraction, and some preliminary observations were made. The examinations are continuing. (TAC 6.14)

Vacuum cathodic etching of as-cast unirradiated U₃Si received from Chalk River reveals a microstructure nearly identical to that reported in AECL-2874. Vacuum heat treatment at 800 °C for 114 hr lowered the hardness of the U₃Si samples from 44.6 Rc to 22.6 Rc and increased the density from 14.70 to 15.18 g/cm³. (TAC 6.15)

The design of the fuel rods and fuel bundle assembly for irradiation in the Whiteshell facility has been completed. A decision was made to employ solid-dished uranium pellets for the unbonded rods rather than providing the internal void space with axial holes. Primary fabrication of Ozhennite coextrusion billet components has been completed.

Primary fabrication of Ozhennite sheath components was completed, and the tube blank extrusion billets were prepared. Tooling for tube blank extrusion was fabricated.

Procurement of adjusted uranium ingots of 2.1, 2.3, and 2.55 wt% ^{235}U enrichment is in progress with National Lead Fernald Plant. Casting of the ingots was completed.
(TAC 6.16)

II. HEAVY WATER REACTOR PROGRAM

A. PROGRAM MANAGEMENT - 12000

Program Management - 12100 (Staff of HWRPO)

Draft reports of HWR status of technology were completed for several areas of technology, including portions of reactor physics, fuel cycle analysis, Zircaloy pressure tubes, calandria, fuel materials, and heat transfer and fluid flow. The information presented in these reports is being used in the preparation of the HWR Program Plan.

Foreign Technical Support - 12200 (Staff of HWRPO)

The HWRPO has assigned a physicist to Chalk River National Laboratory to participate in enriched uranium fuel cycle analyses of mutual benefit to the U.S. and Canadian HWR Programs.

The following information was received from the HWRP representative at Halden:

The shutdown period for loading of test fuel assemblies and plant modifications lasted about one month. The modifications included, in addition to the items described in the previous report, installation of a new purification cooler, installation and checkout of a system for readout of fission gas pressure transducers installed on test fuel assemblies, installation of a regulation valve on the secondary side of the primary sub-cooler, and installation of motors on some smaller valves.

During this period 20 test fuel assemblies and 6 irradiated standard second charge assemblies were loaded. Nine of the test assemblies were irradiated during the second charge operation: IFA-4, 11, 25, 29, 35, 39, 40, 42, and 43. The new assemblies are IFA-101 and 102 (Danish), 106 and 107 (Japanese), 130, 131, 137, and 154 (Italian) and 158 (Swedish).

A list of the test fuel assemblies currently expected to be irradiated during 1967 through 1969 is shown in Table I.

Following a containment test and a leakage test on the top lid, the reactor was started up on November 28, 1967 for calibration and irradiation of the test fuel assemblies.

The IBM 1800 process computer was officially accepted by the Project on September 19, 1967, after completion of installation and hardware/software checkout. Much experience has been gained since that date on the capability of the computer configuration and the reliability of the hardware. Apart from some minor faults, the reliability of the hardware and the quality of the software has been found quite satisfactory. The experience with the computer system indicates that the configuration seems to be well suited for the tasks planned during the present program period.

The computer has been successfully operated on-line during several periods, and significant progress has been made in debugging and checking the initial programs which will be

TABLE I. Survey of 1967 Through 1969 Test Fuel Assemblies

IFA No.	Supplier	Loading Date	Channel Power kW	Material/Enrichment	Fuel Weight, kg	Assembly Type	Cladding Material	Cooling Type	Un-loading Date
101	D ^(a)	Nov 67	530	UO ₂ pellets, 5%	15.0	2 seven pin clusters	Zry-2	Nat.	Aug 68
102	D	Nov 67	530	UO ₂ pellets, 5%	15.0	2 seven pin clusters	Zry-2	Nat.	1969
103	NL	Apr 68	550	UO ₂ powder, 6%	13.0	1 seven pin cluster	Zry-2	Nat.	1969
104	NL	Dec 68	600	UO ₂ powder, 8%	13.0	1 seven pin cluster	Zry-2	Nat.	1969
105	NL	Dec 68	600	PuO ₂ -UO ₂ powder	13.0	1 seven pin cluster	Zry-2	Nat.	1969
106	Ja	Nov 67	550	UO ₂ pellets, 10%	12.0	2 nine pin clusters	Zry-2	Forced	Aug 68
107	Ja	Nov 67	550	UO ₂ pellets, 10%	12.0	2 nine pin clusters	Zry-2	Forced	1969
108	I	Aug 67	0	ThO ₂ -2% UO ₂ pellets	14.0	Zero power burnup assy	Zry-2	Nat.	Sep 67
109	I	Aug 67	240	ThO ₂ -2% UO ₂ pellets	14.0	2 seven pin clusters	Zry-2	Nat.	1969
110	I	Aug 67	240	ThO ₂ -2% UO ₂ pellets	14.0	2 seven pin clusters	Zry-2	Nat.	1969
111	I	Aug 67	240	ThO ₂ -2% UO ₂ pellets	14.0	2 seven pin clusters	Zry-2	Nat.	1969
112	I	Aug 67	240	ThO ₂ -2% UO ₂ pellets	14.0	2 seven pin clusters	Zry-2	Nat.	1969
113	I	Aug 67	240	ThO ₂ -2% UO ₂ pellets	14.0	2 seven pin clusters	Zry-2	Nat.	1969
114	I	Aug 67	240	ThO ₂ -2% UO ₂ pellets	14.0	2 seven pin clusters	Zry-2	Nat.	1969
115	I	Aug 67	240	ThO ₂ -2% UO ₂ pellets	14.0	2 seven pin clusters	Zry-2	Nat.	1969
116	HP	Jan 68	200	UO ₂ pellets, 6%	6.5	2 four capsule clusters	Zry-2	Nat.	1969
117	HP	Jan 68	200	UO ₂ pellets, 6%	6.5	2 four capsule clusters	Zry-2	Nat.	1969
118	HP	Dec 68	200	UO ₂ pellets, 6%	6.5	2 four capsule clusters	Zry-2	Nat.	Mar 69
119	HP	Dec 68	200	UO ₂ pellets, 6%	6.5	2 four capsule clusters	Zry-2	Nat.	Mar 69
120	HP	Mar 69	200	UO ₂ pellets, 6%	6.5	2 four capsule clusters	Zry-2	Nat.	1969
121	HP	Mar 69	200	UO ₂ pellets, 6%	6.5	2 four capsule clusters	Zry-2	Nat.	1969
122	HP	Jan 68	180	UO ₂ pellets, 9.5%	2.0	2 half length pins	Zry-2	Nat.	1969
123	HP	Aug 68	--	--	--	--	--	--	--
124	I	Jan 68	80	UO ₂ -1.8% PuO ₂ vipac	2.0	2 half length pins	Zry-2	Nat.	1969
125	I	Jan 68	350	ThO ₂ -10% UO ₂ pellets	5.3	1 six pin cluster	Zry-2	Nat.	Apr 68
126	I	Jan 68	170	ThO ₂ -10% UO ₂ pellets	2.7	1 three pin cluster	Zry-2	Nat.	1969
127	I	Jan 68	170	ThO ₂ -10% UO ₂ pellets	2.7	1 three pin cluster	Zry-2	Nat.	1969
128	I	Apr 68	250	UO ₂ pellets, 4%	--	2 four capsule clusters	Zry-2	Nat.	--
129	I	Aug 68	--	ThO ₂ -3.3% UO ₂	--	2 four capsule clusters	Zry-2	Nat.	Dec 68
130	I	Nov 67	310	UO ₂ pellets/powder, 6%	8.0	2 six pin clusters	Zry-2	Nat.	1969
131	I	Nov 67	310	UO ₂ pellets/powder, 6%	8.0	2 six pin clusters	Zry-2	Nat.	1969
132	I	Jan 68	100	UO ₂ pellets/powder, 10%	2.0	2 half length pins	Zry-2	Nat.	1969
133	I	Jan 68	100	UO ₂ pellets/powder, 10%	2.0	2 half length pins	Zry-2	Nat.	1969
134	I	Apr 68	--	UO ₂ pellets	3.0	2 four capsule clusters	AlSi 304	Nat.	Jul 68
135	I	Dec 68	150	PuO ₂ -UO ₂ pellets	3.0	2 four capsule clusters	Zry-2	Nat.	1969
136	I	Mar 69	--	--	--	--	--	--	--
137	I	Nov 67	450	UO ₂ pellets, 4%	26.0	3 seven pin clusters	Zry-2	Forced	Apr 68
138	I	Apr 68	450	UO ₂ pellets, 4%	26.0	3 seven pin clusters	Zry-2	Forced	1969
139	I	Apr 68	450	UO ₂ pellets, 4%	26.0	3 seven pin clusters	Zry-2	Forced	Aug 68
140	I	Aug 68	450	UO ₂ pellets, 4%	26.0	3 seven pin clusters	Zry-2	Forced	1969
141	I	Aug 68	450	UO ₂ pellets, 4%	26.0	3 seven pin clusters	Zry-2	Forced	1969
142	I	Apr 68	250	UO ₂ pellets, 4%	--	2 four capsule clusters	Zry-2	Nat.	--
143	G	Cancelled	--	--	--	--	--	--	--
144	G	Cancelled	--	--	--	--	--	--	--
145	G	Apr 68	320	UO ₂ pellets, 9%	5.5	2 four capsule clusters	Zry-2	Nat.	Aug 68

TABLE I. (Contd)

IFA No.	Supplier	Loading Date	Channel Power kW	Material/Enrichment	Fuel Weight, lg	Assembly Type	Cladding Material	Cooling Type	Un-loading Date
146	G	Apr 68	280	UO ₂ powder, 9%	5.0	2 four capsule clusters	Zry-2	Nat.	Aug 68
147	D	Jan 68	550	UO ₂ pellets, 9%	14.0	2 seven pin clusters	Zry-2	Nat.	1969
148	D	Jan 68	550	UO ₂ pellets, 5%	14.0	2 seven pin clusters	Zry-2	Nat.	1969
149	Ja	Apr 68	550	UO ₂ pellets, 10%	14.0	1 nine pin cluster	Zry-2	Forced	1969
150	Ja	Apr 68	550	UO ₂ pellets, 10%	14.0	1 nine pin cluster	Zry-2	Forced	1969
151	G	Aug 68	--	UO ₂ pellets	--	1 six pin cluster	Zry-2	Nat.	1968
152	HP	Aug 68	30	UO ₂ pellets, 2.4%	1.0	1 single pin	Zry-2	Nat.	--
153	HP	Aug 68	30	UO ₂ pellets, 2.4%	1.0	1 single pin	Zry-2	Nat.	--
154	N	Nov 67	300	UO ₂ pellets, 2%	14.5	1 seven pin cluster	Al	Nat.	1969
155	N	Nov 67	230	UO ₂ pellets, 2.4%	6.5	1 six pin cluster	Al	Nat.	1969
156	N	Aug 68	300	UO ₂ pellets, 6%	6.5	1 six pin cluster	Al	Nat.	1969
157	S	Jan 68	450	UO ₂ pellets, 10%	13.0	1 nine pin cluster	Zry-2	Nat.	1969
158	S	Nov 67	370	UO ₂ pellets, 5%	11.0	1 five pin cluster	Zry-2	Nat.	1969
159	Ja	Aug 68	--	UO ₂ -PuO ₂	12.0	1 nine pin cluster	Zry-2	Nat.	1969
160	Ja	Mar 69	--	UO ₂ -PuO ₂	12.0	1 nine pin cluster	Zry-2	Nat.	1969
161	D	Aug 68	--	UO ₂ pellets, 5%	15.0	1 seven pin cluster	Zry-2	Nat.	1969
162	D	Aug 68	--	UO ₂ pellets, 5%	15.0	1 seven pin cluster	Zry-2	Nat.	1969
163	HP	Aug 67	--	UO ₂ pellets, 2.4%	14.0	2 seven pin clusters	Zry-2	Nat.	Sep 67
164	D	Dec 68	--	UO ₂ powder	14.0	1 or 2 seven pin clusters	Zry-2	Nat.	1969
165	D	Dec 68	--	UO ₂ powder	14.0	1 or 2 seven pin clusters	Zry-2	Nat.	1969
166	G	Aug 68	210	UO ₂ pellets, 6%	6	2 four capsule clusters	Zry-2	Nat.	Dec 68
167	G	Aug 68	200	UO ₂ pellets, 6%	6	2 four capsule clusters	Zry-2	Nat.	Dec 68
168	G	Dec 68	200	UO ₂ pellets, 6%	6	2 four capsule clusters	Zry-2	Nat.	Mar 69
169	G	Dec 68	200	UO ₂ powder, 6%	6	2 four capsule clusters	Zry-2	Nat.	Mar 69
170	HP			Halden Corrosion Test Rod (No fuel)					
171	G					2 four capsule clusters		Nat.	
172	G					2 four capsule clusters		Nat.	
173	G					2 four capsule clusters		Nat.	
174	G					2 four capsule clusters		Nat.	
175	I	Dec 68	--	ThO ₂ -3.3% UO ₂	--	2 four capsule clusters	Zry-2	Nat.	Mar 69
176	I	Mar 69	--	ThO ₂ -3.3% UO ₂	--	2 four capsule clusters	Zry-2	Nat.	Mar 69

(a) In the column titled "Supplier," the abbreviations are: D (Denmark), G (Germany), HP (Halden Project), I (Italy), J (Japan), N or NL (Netherlands), and S (Sweden). Blank spaces in the table indicate that the information is not yet available.

operational during power operation of the reactor. About 60 individual on-line programs have been implemented. Among the functions covered by this system are:

- logging, monitoring, and reporting of operational data
- logging, reporting, and calibration of test fuel assemblies
- service programmes for communication units: the operator's console CRT and XY plotter
- preliminary processing of alarms.

Work is in progress on the computer routines which will replace analog controllers. One single loop has been subject to experimental and theoretical study in order to establish simple methods for analysis of individual control loop characteristics. In this connection, the computer has been performing closed loop control of a loop mockup on the Beckman analog computer. In parallel to this, routines have been specified for a uniform and complete treatment of the implementation problems associated with simple direct digital control of 10 individual loops.

A simplified theoretical model of the plant dynamics has been developed, and will be applied in a more sophisticated control system for the vessel pressure. An extensive series of experiments has been performed to verify the model structure and determine parameters in the model. The analysis of these experiments shows that the model generally describes the dynamic behavior of the plant adequately for control purposes, but a small modification is necessary in order to describe the subcooling influence to a sufficient degree of

accuracy. A scheme has been established for rapid evaluation of numerical values for the parameters in the model on the basis of step response techniques.

HWR Information Center - 12300

(B. B. Lane)

A coordinate indexing system was analyzed to determine what application it might have to indexing reports and articles. A number of other forms of index offer advantages not inherent in the coordinate system. Thus, it will not be adopted at the present time.

A chapter of a Russian book on thermal properties of heavy water has been sent to a computing center for translation.

B. GENERAL STUDIES - 13000

Safety Analysis - 13100

(W. R. Lewis)

Review of the status of technology of the safety characteristics of HWR's was continued. With pressure tube HWR's, the inventory of heat in the primary system is generally lower than for LWR's or pressure vessel HWR's, and it is also feasible to use multiple loops. These factors could lead to reductions in the cost of a containment system.

Reactor Physics - 13200

(W. R. Lewis)

The portion of the status of technology report on the kinetics of D_2O

reactors has been completed. It was determined that the problem areas concerning the kinetics of H_2O reactors are very similar and that the technology relative to H_2O reactor dynamics can, for the most part, be readily extended to include D_2O reactors. Some conclusions included in the technology report are:

- 1) Point-kinetics methods are probably not sufficient to describe the dynamic behavior of a large D_2O power plant.
- 2) Solutions to the space-time kinetics equations have been obtained using approximations of varying validity; i.e., finite difference methods, adiabatic model, nodal models, modal models, and flux synthesis methods.
- 3) In one-dimensional situations, the finite difference method, exemplified by the WIGLE codes, is generally accepted to be the most accurate method.
- 4) Two- and three-dimensional calculations are required to fully describe the dynamic behavior of large reactors. However, present day computers cannot solve the finite difference equations for multidimensional situations in reasonable computing times.
- 5) Each of the other approximate methods mentioned above has its relative advantages and disadvantages. However, at the present time, the flux synthesis method appears to be the most promising approach for multidimensional situations.
- 6) Control of xenon oscillations is a difficult problem that also requires two- and three-dimensional analysis. The major goal of such an analysis should be to predict where control rods should be located in order to insure controllability of the reactor. To date, no analysis has been successful in generating control rod design criterion in a suitable manner.
- 7) In order to systematically evaluate the relative merits of all possible approaches to space-time problems, it is appropriate to analyze a typical reactor via all methods. The typical reactor should be a numerical model of a 1000 MWe D_2O reactor.

The portion of the status of technology report in the static physics behavior of D_2O reactors is being completed. Valuable input to the report has been obtained from various individuals at other laboratories throughout the country.

A preliminary draft of the summary, conclusions, and recommendations has been completed for the reactor physics part of the status of technology report.

Economic and Initial Design Studies - 13300

Reactor Studies

(J. R. Young)

A series of preliminary cost analyses for pressure vessel type HWR's was made using the Swedish optimization programs BOP and BIOP. The parameters investigated are listed in Table II with a sensitivity analysis for each.

TABLE II. Sensitivity Analysis HWR Design Parameters

Parameter	Base Value	Change in Value	Change in Unit Cost of Power
Interest rate	7%	Increase to 12%	+41%
Electric capacity	600 MWe	Increase to 1000 MWe	-10%
D ₂ O price	\$20/lb	Decrease to \$13.50/lb	-5%
Plant factor	0.80	Increase to 0.85	-5%
Fuel enrichment	0.711	Increase to 1.25	-5%
Uranium ore cost	\$8/lb U ₃ O ₈	Increase to \$12/lb	+3%
Reactor pressure	1020 psi	Increase to 1520 psi	No change

The most important parameters are the interest rate and the plant capacity, as would be expected because of the high capital costs. On the other hand, the optimum design was relatively insensitive to fuel enrichment, plant factor, D₂O price, ore cost, and operating pressure. These relationships are summarized in Figure 1 for the direct cycle, natural circulation reactor.

In general, the direct cycle, natural circulation reactor had a lower power cost than the indirect cycle and forced circulation concepts for the economic assumptions used.

The study is an excellent example of how (adverse) changes in single parameters can be compensated for by reoptimizing the reactor plant. It illustrates the importance of optimizing reactor plants when changes have been made in economic or engineering aspects of the plant. A topical report containing more detailed information on the results will be written.

Fuel Cycle Studies

(G. A. Last)

The status of technology for HWR fuel cycles was completed. Reviews were made of studies conducted by Oak Ridge National Laboratory, Savannah River Laboratories, and Pacific Northwest Laboratories; Atomics International-Combustion Engineering, Babcock and Wilcox Co.; and Canadian studies representing work by the Canadian General Electric Co. and Atomic Energy of Canada Limited. The objectives of the review were to summarize procedures and techniques used and significant results obtained, and to determine requirements for further studies for evaluating HWR concepts. The conclusions developed from this study are summarized in the following paragraphs.

Because of differences in economic ground rules and assumptions, direct comparisons between fuel cycle and total power costs for all the HWR evaluations from different studies

FIGURE 1. Effect of Reactor Design Parameters on PHWR-PV Power Costs

are impossible. The fuel cycle analysis data summarized in this study should be normalized to a consistent and up-to-date reference set of ground rules to provide a basis for making more direct cross comparisons.

Because of the heavy emphasis on HWOCR development during the past few years, much of the recent analysis data in the U.S. literature relates to HWOCR evaluations. Because of similarities in nuclear properties between HWOCR's and PHWR's much of the HWOCR data can be applied to PHWR evaluations. A close to optimum oxide-fueled PHWR design can be defined by extrapolating results from HWOCR studies. This would be a first step in providing an up-to-date evaluation of PHWR's. (See following section on HWR Fuel Cycle Study).

The difference between equilibrium cycle and 30-year levelized fuel cycle costs was found to be quite small (generally not more than 5%). The more straightforward equilibrium cycle costs can be used in preference to 30-year levelized costs in design optimization studies; 30-year levelized cost procedure should be used for final calculations.

Selection of economic ground rules can have as much or more of an impact on the results of reactor evaluation studies than almost any other factor. A significant effort will have to be concerned with development of ground rules. The ground rules should recognize the existence of competing LWR's and the

effects of market penetration rate on HWR costs.

In optimizing a HWR design it is very important to consider the entire system because of cost trade-offs between fuel cycle optimization thermal-hydraulics optimization, and other aspects of the reactor plant (see previous section on reactor studies). The procedures and techniques used by SRL, AI-CE, and CGE for optimizing HWR designs should be thoroughly evaluated, and consideration should be given to expanding the flexibility of these procedures and to automating the entire optimization procedure.

Reported fuel cycle costs often indicate accuracies, in terms of significant figures, beyond that justified by the accuracy of the input data. To provide a more objective evaluation of results, a statistically sound evaluation procedure should be developed for the HWR evaluation studies and incorporated as part of the routine procedures.

Several heavy water reactor design concepts have been the subject of a broad based evaluation and cost normalization procedure during the current AEC Civilian Nuclear Power Study. These data should be used as the starting point for preparing up-to-date comparative evaluations of the HWR concepts of interest.

To provide for continually improving accuracy in the physics codes used for burnup calculations, a continuing program should be established to compare analyses of spent fuel discharged from operating HWR's

with the results of codes used for burnup calculations.

To aid in understanding the leverage that the primary variables have on fuel cycle costs and to provide guidance for focusing R&D effort, a set of cost correlations for the major variables in each HWR concept of current interest should be developed using a consistent set of ground rules.

HWR Fuel Cycle Study

Because of similarities in the nuclear properties of HWOCR's and PHWR's, and because of the cost normalization procedures that have been applied to the HWOCR designs being evaluated in the current AEC Civilian Nuclear Power Study, the HWOCR designs are considered to be the best starting point to develop up-to-date power costs for near optimum PHWR's. For these reasons, conversion of the HWOCR designs to PHWR designs is being undertaken. Appropriate corrections will be applied to the capital and fuel costs to account for differences in number of pressure tubes, fabrication costs, thermal efficiency, etc.

The first step in this conversion procedure has been an effort to match the UC fueled HWOCR material balance data with an identical reactor design using the ALTHAEA burnup code. Once this match is completed, a PHWR case will be run to determine how well the assumption of similar

nuclear properties for the two concepts checks out. However, because of recent revisions in the cross-section library, the first attempt at matching the HWOCR data has been unsuccessful, and the proper calibration of the ALTHAEA constants has not yet been obtained.

Heavy Water Studies

(J. R. Young)

The effect of heavy water costs on HWR unit power costs is being investigated to determine the incentives for R&D efforts to reduce heavy water costs and inventories. Typical inventory data for operating and proposed reactors are plotted in Figure 2. The wide variation of 0.2 to 0.7 tons/MWe (or 0.13 to 0.46 mills/kW-hr) in large size reactors appears to result primarily from differences in fuel enrichment and coolant types. In HWR's with separate primary and moderator-reflector heavy water systems, the primary system inventory is typically 20 to 25% of the total. The incentive for reducing inventory size by one-quarter and the heavy water unit cost by one-half can be illustrated in a preliminary fashion by analysis of unit power costs for two 1000 MWe reactor plant studies. Table III summarizes the analysis. The analysis indicates that savings up to \$2,000,000 per reactor per year would result. This corresponds to about a 7% reduction in unit power costs.

FIGURE 2. Heavy Water Inventory for Reactors

TABLE III. Impact of Heavy Water on Power Costs for 1000 MWe HWR's

	Reactor Type	
	PHWR ⁽¹⁾	HWBLW ⁽²⁾
D ₂ O Inventory, tons	680	545
Annual D ₂ O Loss, tons (0.5%/yr)	3.4	2.7
D ₂ O Inventory Value (\$17.50/lb)	\$23,800,000	\$19,070,000
Annual Costs		
D ₂ O Inventory	\$3,140,000	\$2,520,000
D ₂ O Loss	\$119,000	\$95,000
Effect on unit power costs, mills/kW-hr		
D ₂ O Inventory	0.45	0.36
D ₂ O Loss	0.02	0.01
Benefit from 50% lower D ₂ O unit cost and a 25% smaller inventory:		
Annual cost	\$2,040,000	\$1,630,000
Unit power cost, mills/kW-hr	0.29	0.23

(1) J. M. Boswell, et al. "Heavy Water Moderated Power Reactors Cooled by Liquid D₂O or H₂O," DPW-64-135. March 29, 1964.

(2) Preliminary data from Advanced Converter Task Force.

C. RESEARCH AND DEVELOPMENT STUDIES - 20000 thru 99000

Reactor Vessels - 21100

(R. P. Marshall)

A review of the technology of prestressed concrete reactor vessels (PCRVS) indicate that two major uncertainties for water reactors relate to the long-term behavior of insulation and the behavior of PCRVS's during maximum credible incidents.

To define these and other uncertainties in greater detail for HWR application, it appears necessary to study a HWR in a PCRVS configuration. It has been suggested by personnel in the PCRVS field that nuclear envelope data from previous large-sized HWR designs should provide adequate models. Such a study should permit the uncertainties in each phase of the design, construction, and operation of a PCRVS-HWR to be itemized and evaluated.

Pressure Tubes - 21300

(R. P. Marshall)

A report summarizing the status of Zircaloy pressure tube technology has been drafted.

The HWRPO pressure tube program has been formulated with the major objective of qualifying Zircaloy pressure tubes as fully tested HWR components. This program is based on the behavior needed for long reactor service, the possible modes by which a pressure tube could fail mechanically or behaviorally, and the technical base presently available in each area. The critical areas and, therefore, the areas of principal investigation include:

- Growth characteristics of flaws (fret marks, etc.) to critical size for fracture
- Crack propagation in thick-walled geometries
- Creep and irradiation growth under complex stress systems
- Strain to failure under complex stress systems
- Criteria and techniques to determine approach to failure for reactor surveillance.

Other important subjects will include transition joints, fabrication development, corrosion and hydrogen pickup, uniaxial mechanical properties, and the development of advanced materials and designs such as the composite and the internally insulated designs.

Two major advancements will be the development of Criteria for Pressure Tube Design, to utilize the beneficial

effects of the thick-wall geometry and a favorable crystalline texture, and Criteria for Pressure Tube Removal, to establish in-reactor surveillance program and techniques and to instruct the HWR operator when a pressure tube should be removed because of its behavior or condition.

Zircaloy was selected for qualification because of the large technical base presently available (approximately 1800 Zircaloy pressure tubes are presently installed in nuclear water reactors), and because it is not apparent at this time that other materials have better properties in all critical areas and, therefore, what the property trade-offs might involve. During the several years required to perform the pressure tube program, work in Canada, U.S., and U.K. will provide much data for similar qualification of Zr-2.5% Nb tubes.

The status of engineering standards and specifications for the material, fabrication, inspection, and reactor surveillance of Zircaloy pressure tubes is being reviewed. These standards and specifications will be classified to show:

- Areas where existing standards and specifications are adequate
- Areas where existing standards and specifications should be modified
- Areas where no standards and specifications exist or where they are inadequate.

These compilations are required to define where additional R&D is necessary.

Calandria - 21500

(R. P. Marshall)

Review of the status of technology for HWR calandria assemblies indicates that both aluminum and zirconium alloy-stainless steel types are used and have performed satisfactorily in nuclear service. No major problems are expected in the design, fabrication, and performance of calandrias for 1000 MWe HWR's. The following uncertainties were identified as areas where development work may be needed:

- The primary materials problem relates to welding during fabrication, particularly for large vessel sizes where substantial field welding (and field inspection) may be necessary. In HWR sizes of about 1000 MWe or larger it may be necessary to largely field fabricate and inspect the calandria assembly.
- Design of the supporting structures for large HWR's could be very difficult in view of the vessel size and the complexity of primary piping systems.
- Moderator dump systems may be unsuitable for large size reactors because reactivity removal rates may not be sufficiently rapid. This aspect requires further analysis.

Fuel Materials - 28100

(G. A. Last)

Uranium and Thorium Metallic Fuels

Reviews of the status of technology of uranium- and thorium-base fuels and

fuel elements have been completed. It is concluded that exposure limits for tubular uranium elements are greater than 10,000 MWd/T, those for hollow-core uranium elements beyond 20,000 MWd/T, and those for tubular thorium elements beyond 50,000 MWd/T.

These conclusions are based on the observed performance of fuel elements irradiated in water-cooled reactors, the swelling behavior of test fuels irradiated under conditions applicable to heavy water reactors, and basic swelling studies.

The following were defined as areas of work required to fully evaluate the potential of these types of fuel for HWR application:

- 1) Confirm that tubular uranium elements irradiated under HWR conditions to exposures greater than 10,000 MWd/T will perform as satisfactorily as do shorter tubular elements of uranium.
- 2) Establish the operational characteristics and limitations of clusters of hollow-core uranium rods irradiated under HWR conditions to burnups greater than 20,000 MWd/T.
- 3) Develop behavior-exposure relationships for tubular thorium fuel elements irradiated to at least 50,000 MWd/T under HWR conditions so that reasonable estimates of the probable limitations of this type of fuel can be made.

Ceramic Fuels

Ceramic fuel technology has reached an advanced state: two-thirds of all the operating power reactors (light water, heavy water, gas-cooled

and liquid-metal-cooled) use ceramic fuel, predominantly UO_2 . However, fuel operating limits are not yet fully defined, and much information is still required to define limits and optimum fuel geometries. Much reactor fuels data from light water reactors are applicable to heavy water reactors.

Conclusions made from the status of technology survey include the following:

- 1) Ceramic fuels in rod assemblies are almost universally accepted. A trend toward larger fuel assemblies is seen. Future planning emphasizes the use of rod bundles, but four U.S. studies have examined the use of tubular elements. The use of tube-in-shell elements has been considered in Canada, but use of this fuel geometry is not presently planned for any HWR.
- 2) The effects of fission products and fuel stoichiometry on the performance of fuel rods under high power level operation and high-burnup conditions needs better definition.
- 3) Central core melting is generally indicated to be the limiting factor for power reactor ceramic fuels. However, the feasibility of fuel element operation with partially molten core material should be established.
- 4) The maximum linear rod power of ceramic fuel rods for heavy water power reactors in operation or under construction is in the 14 to 19 kW/ft range. It appears feasible to increase the maximum rod power to 21 to 23 kW/ft with con-

tinued development and evaluation studies. There is some evidence that the behavior of defective elements may become a limiting factor if the allowable maximum rod power is raised above approximately 23 kW/ft.

- 5) Zircaloy is used as the cladding material for most LWR fuels, although stainless steel may have application. There is very little indication that materials other than zirconium base alloys will find application as cladding materials for HWR's.

A need for Ceramic fuel development studies is indicated in the following areas in order to advance fuel technology and/or to improve the competitiveness of HWR's:

- 1) Irradiation testing of fuel at substantially higher linear rod power ratings: e.g., above 21 kW/ft and particularly above 27 kW/ft
- 2) Irradiation testing of defect fuel element under high linear heat rating conditions
- 3) Evaluation of fuel and fission product migration, fuel slumping and fuel swelling relationships with temperature, stoichiometry and burnup
- 4) Evaluation and irradiation testing of radial fuel enrichment within a fuel assembly

Heat Transfer and Thermal Hydraulics - 28900

(G. A. Last)

The survey of the status of heat transfer and fluid flow technology as applied to heavy water reactors was

completed, and a preliminary draft of the final report was prepared. This work was directed toward identifying technology development requirements for the design and operation of a 1000 MWe heavy water reactor of the pressure tube (HWR-PT) or pressure vessel-type (HWR-PV). As a result of this survey, it was concluded that the future development of heavy water reactors will not be limited by the lack of thermal-hydraulic technology if the following action is taken:

- 1) Present studies in the light water reactor field should be expanded to include configurations and conditions more applicable to heavy water reactors. In most cases, only a slight expansion of present programs is necessary. Although there is research in every field of thermal hydraulics that could be included, the areas that require the most attention relate to reactor transients and emergency cooling. In these areas, there are differences between pressure-tube and pressure-vessel reactors which must be studied and evaluated. Specific areas are (1) studies of loss of coolant from pressure-tube reactors and (2) studies and evaluations of emergency core cooling schemes to cover special problems encountered with pressure-tube type reactors. Presumably, a typical maximum credible accident (MCA) would be similar to some LWR MCA's: e.g. failure of a main coolant header. However, the feeder piping and discrete coolant channels introduce differences between the pressure tube and pressure vessel concepts.

- 2) Several specific areas of non-accident oriented heat transfer and fluid flow that require further work include (1) the importance of fuel geometry on boiling and nonboiling heat transfer, and, in particular, the effect of rod spacing and the importance of mixing in a multirod bundle (2) the interaction between coolant flow instability and core heat transfer, and, although the importance of instability on the critical heat flux has been shown, the interrelationship with coolant instability.

Coolant Technology - 35100

J. A. Ayres

A status report for HWR coolant technology is being prepared for the HWR Program Plan. A summary is presented here.

The rates and mechanisms of corrosion in D_2O are almost identical to those in H_2O . Therefore, there are no corrosion problems unique to HWR's. However the consequences of corrosion in HWR's are potentially more serious since the loss or degradation of large quantities of D_2O could be very expensive. Large HWR's would have D_2O inventories ranging in cost from \$15,000,000 to \$40,000,000.

The coolant chemistry (pH, additives, quality control) will be essentially the same for HWR's and LWR's, although HWR's may be somewhat more restrictive. For example, the coolant in N Reactor, a PWR, is maintained by feed and bleed, whereas in any HWR a closed system must be used for purification. Additional instrumentation

will be needed to check possible sources of loss of coolant. If a reactivity poison is added, methods for removing the poison without loss of coolant must be provided.

If a reactor is properly designed, corrosion per se will very seldom be a problem. Uniform corrosion rates of carbon steel, stainless steel, Inconel, and Stellites are so low that during the lifetime of the reactor, the average penetration will be less than 0.001 in. Other corrosion resistant materials are available for fabrication of minor components. Nonuniform corrosion (galvanic, cavitation, pitting, cracking) can be reduced to insignificance by proper design. A system containing both carbon steel and stainless steel will have many galvanic junctions. During normal operation, these will not cause trouble. Care should be taken during off-standard operation (shutdown, chemical cleaning) to ensure conditions are not too corrosive; oxygen content should be as low as possible; stagnant concentrated solutions should not remain in contact with galvanic junctions for long periods of time. Concentrated chemical solutions, such as may be present during chemical cleaning or by concentration of chemicals normally in the coolant, may cause localized corrosion of carbon steel, stainless steel, and Zircaloy. Concentrated LiOH solutions are very corrosive to Zircaloy at high temperatures; concentrated NaOH can lead to caustic cracking; concentrated sulfates can cause pitting or rapid corrosion of carbon steel, stainless steel, and

Inconel; chlorides and oxygen are very corrosive and lead to cracking of stainless steels.

Characteristically, reactors are seldom designed adequately from a corrosion standpoint, and problems of nonuniform corrosion exist. What is needed is some reliable instrument for continuously monitoring corrosion in the operating system: an instrument similar to that used for monitoring of corrosion in the oil industry. Several methods have been suggested: measurement of polarization or changes in resistance to monitor uniform corrosion, galvanic attack, or crevice attack of carbon steel, stainless steel, or Inconel; measurement of zirconium concentration in water to monitor Zircaloy fretting. So far, results have been negative. There are no proven methods for monitoring corrosion.

Although corrosion per se is usually not important, the corrosion products formed can cause serious trouble. If a reactor is operated routinely, the corrosion products will deposit almost 100% on the piping surfaces. The deposition coefficient is very much higher than the release coefficient, and the purification bypass removes only a very small amount of the corrosion products formed in the system. Even though corrosion rates are low, the large surface area and large volume of water produce a large volume of corrosion products that can cause two types of problems: (1) deposition in narrow passages, leading to flow restrictions, and (2) deposition of activated corrosion products in the piping systems leading to restricted

access for normal operations and maintenance.

The greatest difficulty so far has resulted from deposition of radionuclides. The activity levels build up with time. Since the isotopes, especially ^{60}Co , have a long half life, equilibrium is not attained for a very long time, and levels can be restrictive.

The rate of radioisotope buildup and final activity level depend on the operating variables, water treatment, and materials of construction, but the important variables are unknown: how each affects radioactivity buildup, how they can be manipulated to keep activity levels down, or how to choose materials of construction intelligently. Two examples will suffice: (1) Most of the radioactivity in the deposits is due to ^{60}Co , which has a very long half life (5.2 year) and a very hard γ . Yet designers use cobalt alloys for valves and seals; (2) CuNi, used for feedwater heaters in some direct cycle reactors formed films that were most difficult to remove. It also deposited in the in-core region. Usually, these types of problems are learned from experience. It is not always possible to predict what problems will occur or how serious they will be.

Once the radioactive films are formed, the choice must be made to: (1) operate with high activity levels in the system, (2) clean parts of the system, or (3) clean the entire system. In LWR's activity levels increase linearly with time as is shown in Figure 3. The levels may not be sufficiently high to seriously affect

normal maintenance until after 5 to 10 years, but they will eventually present a problem when it becomes necessary to enter the system or remove some component.

The most effective method for dealing with high contamination levels is to clean the entire system, but no reliable, safe, tested procedure is available for most reactors operating today. Furthermore, reactors are not designed to be decontaminated, because traditionally, designers know little about contamination or decontamination. The procedure usually followed after a reactor becomes contaminated is illustrated by the method used at one reactor, where only the components or areas with highest activity levels were decontaminated. The procedure required a large number of personnel to take high radiation levels, and even then only a very small fraction of the primary coolant system was decontaminated. The end result was inefficient use of personnel and longer downtime. Furthermore, the radionuclide inventory in the primary system remained high and provided a source of subsequent contamination of the cleaned components.

Reactor decontamination experience is very limited. Decontamination should be considered a part of the operating cycle, but the knowledge to specify a satisfactory procedure is lacking. Procedures have been developed to decontaminate some reactors (Shippingport, PRTR, N Reactor), but they are expensive, time consuming, and potentially dangerous (corrosive). The reagents are those one would like

FIGURE 3. Shutdown Radiation Levels, Long-Lived Activity
Primary Coolant Recirculation Pipes

to keep out of the reactor system [NaOH, H_2SO_4 , $\text{NH}_2\text{SO}_3\text{H}$, $(\text{NH}_4)_2\text{HC}_6\text{H}_5\text{O}_7$].

These statements are even more pertinent for heavy water reactors. When PRTR was decontaminated, over 90% of the time was required for preparation and restoration (removing fuel elements, removing D_2O , filling with H_2O , removing H_2O , filling with D_2O , replacing fuel elements). The loss of D_2O represented a considerable expense, and operating errors could have resulted in degradation of large quantities of D_2O . While these considerations are not of major importance for a test reactor, where expense of operation is not of over-riding importance, it could not be tolerated in a power reactor if the reactor is to remain competitive. New procedures are required for all water-cooled reactors and especially for HWR's. Some cyclic procedure should be developed to permit decontamination either (preferably) during operation or quickly during shutdown. High-temperature carbon steel systems in the fossil fuel industry use procedures for continuously removing films or preventing films from forming. Some of these procedures may be adapted to nuclear reactors.

Very little is known about the types of films formed in nuclear reactor systems under different conditions, the effects of variables or construction materials on these films, or the relationship between structure and ease of removal. Almost nothing is known about the chemical reactions occurring during decontamination or the effects of chemicals on the films. Procedures have not been developed in an orderly manner. Many reagents are tried until an effective one is found for one particular system. The result is that the reagents that were apparently successful for Shippingport are ineffective for PRTR or Dresden; procedures for PRTR are ineffective for Dresden. A concern is that some modification of a reactor system (e.g., changing from stainless steel to Inconel tubes) might cause changes in the film and make it difficult to decontaminate all or part of the reactor.

The ultimate for any reactor, especially for a Heavy Water Reactor, is a procedure so safe and simple that when activity levels become too high, the reactor can be decontaminated in a routine and efficient manner with very little if any loss of operating time and with no deleterious effects on the reactor.

III. USAEC/AECL COOPERATIVE PROGRAM ON HEAVY WATER REACTORS

A. HEAT TRANSFER AND FLUID DYNAMICS

Subchannel Code Development and Local Subchannel Analysis of Burnout Data - TAC 3.1.6

Pacific Northwest Laboratory
(D. S. Rowe and F. R. Zaloudek)

In FY 67, the computer code COBRA was developed at PNL to enable the analyses of local coolant conditions in rod bundle fuel elements by considering the effects of turbulent mixing and diversion cross-flow between the coolant subchannels during single and two-phase conditions. This code was developed to enable the calculation of local enthalpy in rod bundle fuel elements for the analysis and correlation of existing boiling burnout data. Improvements and modifications are now being incorporated in the original formulation of the code. Already completed are changes designed to consider the following:

- Variable rod gap and channel area along the bundle length and different hydraulic characteristics in the individual channels
- Subcooled void formation
- Thermal conduction in the fluid.

Also, the capability to use optional subcooled and boiling mixing correlations was incorporated into the code.

Using this modified code, a sample problem was computed, and the results were compared with those obtained with the earlier version of COBRA. Agreement was observed verifying the functional operation of the code.

Calculations are continuing to check out the various options available to the program user.

Reports have been received from various users of the original version of COBRA that failures of the subroutine CROUT5 were experienced when certain types of problems are considered. The purpose of the subroutine CROUT5 is to solve a system of simultaneous equations generated during the solution of the problem. It was subsequently found that the mathematical method used in this subroutine is subject to problems when "ill-conditioned" equations are considered. An improved subroutine is being developed to circumvent these problems.

Critical Heat Flux in a 9-Foot Bundle - TAC 3.3.8

Columbia University Engineering
Research Laboratory
(J. E. Casterline)

During the experimental operation of the 9' Uniform Bundle with Simulated End Plates (described in the October and November, 1967 Monthly Progress Letters) some data were taken beyond the critical heat flux limit. Generally, power was added in steps of 0.1 MW after CHF was reached until a rod temperature of 900-1000 °F was indicated. On Figure 4, the maximum surface temperature measured on any of the outer rods is plotted on a graph of outer rod heat flux versus average exit steam quality. This

FIGURE 4. Maximum Rod Surface Temperature Beyond Critical Heat Flux for the 9 ft-18 Rod, Uniformly Heated, Segmented Bundle

graph shows that the effect of a unit increase in power is much more severe at a mass velocity of $0.5 \times 10^6 \text{ lb/m ft}^2$ than at 1.0 or 1.5×10^6 .

Visual inspection of the rods after testing indicated that crud deposition was greatly accelerated on the heat transfer surfaces which had been operated beyond CHF.

Measurement of Flow and Enthalpy in the Subchannels of a BLW-250 Geometry Rod Bundle - TAC 3.3.9
Columbia University Engineering Research Laboratory
(J. E. Casterline)

The assembly of the test section, described in the Monthly Progress Letter for November, was completed. Experiments will be conducted in the early part of January.

HWR 28-Rod Bundle - TAC 3.3.12
Columbia University Engineering Research Laboratory
(J. E. Casterline)

Additional Power

The USAEC, New York Operations Office, has requested the following additional information related to the purchase of additional power generating equipment:

- 1) Additional justification for the selection of the proposed vendor.
- 2) Technical information including descriptive material of the proposed installation and site.

- 3) Type of performance guarantees that will be supplied by the vendor.

This information is being prepared, and will be submitted to NYOO early in January.

B. STABILITY AND CONTROL

Analysis of Spatial Flux Control - TAC 4.3

The report for December was not received in time to be included in this report.

C. REACTOR PHYSICS

Temperature Coefficient Measurements - TAC 5.2

The report for December was not received in time to be included in this report.

D. FUELS AND MATERIALS TECHNOLOGY

Evaluation of Zr-2.5 wt% Nb Pressure Tubing - TAC 6.9

Pacific Northwest Laboratory
(P. J. Pankaskie)

The general objective of this program is to evaluate Zr-2.5 wt% Nb alloy tubing as a pressure tube material with reference to tests and reactor experience that have been obtained on Zr-2 pressure tubes.

In PRTR Test 125, a Zr-2.5 wt% Nb alloy pressure tube (0421) is being

irradiated in the PRTR. This pressure tube is operated as a standard fuel channel except that the operating hoop stress is 19,000 psi instead of about 12,500 psi for a standard Zr-2 pressure tube. From PRTR Test 125, it is expected that both in-reactor creep data and general reactor service data will be obtained. This tube has now reached an integrated exposure of about 2 by 10^{20} nvt. Arrangements are being made for diametral measurements in February of 1968 during an extended reactor outage.

In PRTR Test 127, four crack propagation test specimens are being irradiated in the PRTR. Of the four test specimens, two are hydrided and two are in the non-hydrided, as-received, condition. In November the test assembly was removed for inspection of the test assembly and the Zr-2 pressure tube in which it is being irradiated. Both were found to be in satisfactory condition. In recharging, damage to the test assembly holder occurred. Repairs have been made and the test assembly recharged into the reactor.

Three fatigue crack growth and crack propagation tests with multiple clustered defects have been completed. In these tests two EDM (electrical discharge machined) slots were placed parallel to each other and parallel to the axis of the tubular test specimen. All EDM slots were as identical as possible and each was 10 mils or less in width and with a radius of 0.150 in. In all three tests circumferential separation was 3/8 in.;

axial separation ranged from zero to 0.6 in. All were cyclically pressurized to a peak tensile hoop stress of about 23,000 psi and a minimum of about 1000 psi. For two of the three specimens, the fatigue cracks leaked when the crack length, measured at the OD surface, reached 0.43 and 0.47 in. In the third test, leakage occurred when the fatigue crack had reached a length of 0.30 in. as measured at the OD surface. In this test leaking occurred at 2250 cycles, which is less than half the fatigue cycles for leakage in the other two test specimens. Because of the short crack length and low number of fatigue cycles at which leaking occurred, it is suspected that the EDM slot may have been accidentally placed in an area where the tube wall was thinner than was the case for the other two test specimens. Further analysis will be done to determine the cause for the difference in test behavior.

The use of acoustic emission in zirconium alloy pressure tube evaluation has progressed with the demonstration that it is an extremely sensitive method of detecting micro and macro plastic strain in tensile, fatigue, and burst tests.^(1,2) A solid state electronic readout system to minimize spurious signal and to obtain better pulse character presentation has been successfully designed, fabricated and used in the laboratory. Additional solid state cards to provide automatic pulse rate and pulse energy chart presentations are in the prototype stage. Work will now concentrate on correlation of the presence of flaws and the

crack generation and propagation characteristics of zirconium alloys with their acoustic emission.

References

1. T. E. Michaels and M. C. Fraser. Acoustic Emission Behavior of Zircaloy-2 Pressure Tubing Under Applied Stress, BNWL-545. September 1967.
2. T. E. Michaels. Acoustic Emission of Zircaloy-2 During Tensile and Fatigue Loading, In preparation.

In-Reactor Measurements of Creep in Zr-2.5 Nb Alloy - TAC 6.10

Pacific Northwest Laboratory

(E. R. Gilbert)

An unirradiated control creep test was conducted on Zr-2.5 Nb at 400 °C and 31.6 kg/mm². The measured creep rate was 1.1×10^{-4} hr⁻¹ and the rupture time was 318 hr. The onset of tertiary creep occurred at approximately 2% strain.

In-Reactor Corrosion of Zirconium Alloys - TAC 6.14

Pacific Northwest Laboratory

(A. B. Johnson, Jr.)

X-ray diffraction patterns were obtained from six zirconium alloy coupons irradiated eight cycles (1.7×10^{21} n/cm², >1 MeV) in the G-7 loop of the Engineering Test Reactor. Weight gains, hydrogen analyses and metallographic examination of the 8-cycle specimens appeared in recent Monthly Progress Reports. One coupon from each of six zirconium alloys was analyzed with the oxide in situ. The

irradiated specimens were analyzed on a shielded X-ray spectrophotometer, using a copper target. The following unirradiated controls also were analyzed: prefilmed (13-18 mg/dm²) specimens of Zircaloy-2 and Zr-2.5 Nb; a heavily-oxidized (990 mg/dm²) Zircaloy-2 specimen; an unoxidized coupon of crystal bar zirconium.

Summarizing the results:

- The diffraction patterns on irradiated and unirradiated specimens were basically similar. The major monoclinic line was present in all patterns, but several monoclinic lines were missing from the patterns from in-reactor and out-of-reactor specimens. The missing lines are almost certainly due to orientation effects, since monoclinic oxide is expected on zirconium alloys under the out-of-reactor prefilm conditions (400 °C, 1500 psi steam).
- Strong metal lines occurred on patterns from the unirradiated controls and on in-flux specimens having thin oxides (20-40 mg/dm²). The metal lines were faint or absent on specimens with thick oxides (up to 862 mg/dm²).
- Oxide line intensities increased upon irradiation due to thickening of the oxide.
- Faint lines corresponding to the strongest cubic-tetragonal lines occurred in the irradiated Zircaloy-2 and Zr-Cu-Fe patterns. Collection of oxide for powder diffraction analysis will be attempted, to confirm the presence of the phase and to differentiate between the cubic and tetragonal

structures if either is present.

- There are indications of line broadening for metal and oxide, but the analysis is not complete for all alloys.

Ozhennite and Zircaloy-2 specimen preparation is underway for specimens to be charged into the G-7 loop. A specimen holder is scheduled for charging early in February 1968.

U₃Si Irradiation - TAC 6.15
Pacific Northwest Laboratory
(R. D. Leggett)

The purpose of this program is to establish the dependence of swelling in U₃Si on temperature, pressure, burnup, burnup rate, and metallurgical state. Small specimens will be irradiated in temperature and pressure controlled NaK filled capsules and the changes in density, hardness, and structure evaluated by detailed post-irradiation examination. Sections of fuel rods which have been or will be irradiated by AECL will be examined in order to adjust the present examination techniques to U₃Si fuel material and to allow cross-comparison of results.

Irradiation Program

Calculations are being made to determine enrichment and heat transfer parameters for the irradiation of four U₃Si specimens in each of three modified temperature-pressure controlled Swelling Program capsules.

Postirradiation Examination

It is anticipated that sections from three irradiated U₃Si rods will be shipped to PNL shortly for post-irradiation examination. In the interim, unirradiated samples of U₃Si material prepared at Chalk River have been received and are being subjected to heat treatment, density, hardness, metallographic and X-ray diffraction studies in order to familiarize PNL with this material.

The samples consisted of 3 pieces 13/16 in. diameter by 2 5/8 in. long. Specimen No. 3-2 was cut into six 1/2 in. long pieces. These have been processed as follows: Specimen 3-2 A and 3-2 F are held for historical reference, 3-2 C for as-cast metallography, and 3-2 B, 3-2 D and 3-2 E were deltized at 800 °C for 114 hr in vacuum. Specimen 3-2 D is being prepared for metallography. Mechanical polishing followed by vacuum cathodic etching revealed a microstructure in the as-cast specimen nearly identical to that reported in AECL 2874.

The deltizing heat treatment produced the following changes in the properties of the material:

	Hardness, Rc	Density, g/cm ³
As-cast	44.6	14.70
Deltized	22.6	15.18

These results agree reasonably well with those reported in the literature. The density of U₃Si is reported to be 15.58 g/cm³.

Future work will consist of examination of the heat treated material followed by heat treatment of a larger specimen to determine the degree of dimensional change caused by the del-tizing heat treatment.

Uranium Alloy Fuel - TAC 6.16
Pacific Northwest Laboratory
(R. D. Leggett)

The objective of this program is to develop the design data required to fabricate uranium fuel elements capable of operating at power ratings and/or coolant temperatures high enough to require fuel temperatures in the 500-700 °C range to average exposures of about 7000 MWd/T. The basic fuel design utilized will be bonded and unbonded rod cluster elements with the individual rods containing internal void space to accommodate fuel swelling by internal expansion without straining the clad and increasing rod dimensions. Personnel at the Pacific Northwest Laboratory will fabricate the fuel and personnel at the Whiteshell Nuclear Research Establishment will irradiate the elements in the WR-1 reactor.

Fuel Element Design

The final design of the fuel rods and bundles for irradiation in the Whiteshell facility has been completed and detail drawings of the components have been reviewed. The rods will be 0.600 in. OD and the bundles will contain twelve fuel rods in an outer ring of 2.500 in. pitch diameter and six

dummy rods in an inner ring of 1.300 in. pitch diameter. Each bundle will contain bonded and unbonded rods, each with 0.020 and 0.035 in. thick Ozhennite cladding and 0, 5, and 10% void volumes in the core. The core material will be adjusted uranium with enrichment levels of 2.1, 2.3, and 2.55 wt% ²³⁵U for the different void volumes. A decision was made to provide the void volume in the unbonded pellet fuel by machining dished ends on solid pellets. The void volume in the coextruded fuel will be provided with an axial hole introduced during coextrusion.

Fuel Element Fabrication

Portions of the Ozhennite zirconium alloy primary extrusions were machined to billets for tube blank extrusion. Tooling for these extrusions was designed and fabricated. These tube blanks, 0.780 in. OD by 0.650 in. ID, will be reduced to 0.601 in. OD (0.020 and 0.035 in. wall) tubing by drawing. Drawing dies for this work were received and mandrels are on order.

Billet design for the coextrusions was completed and drawings prepared for machining of components. Primary extrusions of Ozhennite 2.25 in. OD by 1.750 in. ID will be machined for the cladding components.

Adjusted uranium containing 100-150 ppm silicon, 300-400 ppm iron, 700-900 ppm aluminum, and 300-750 ppm carbon is being prepared at the National Lead Fernald Plant as 5.85 in. diameter by 15 in. long solid beta heat treated billets. Alpha phase primary

extrusion of these billets will provide stock for the coextrusion billets. Casting of the uranium ingots was completed. Two of the ingots had satisfactory top and bottom chemistry; one required remelting due to low (80 ppm) silicon content. Receipt of this stock is expected in early January.

Tooling is being designed for positioning and locating supports for attachment to both the sheath tubing and coextruded fuels. These will be

attached by magnetic force welding. Tooling is also being designed for stamping contoured support tabs of Ozhennite with projections for welding. The end cap design and welding procedures are being reviewed.

E. D₂O TECHNOLOGY

D₂O Technology - MIT - TAC 7.2

Reports in this activity are planned on a quarterly basis.

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