BNWL-840

UNCLASSIFIED

DECLASSIFIED by CG-NMP-2, 9/00 and
Approved for Public Release
Name/Date Buzz Hammer 5-15-03
Name/Date W.F. Nicaise, 5-16-03
ORG: PNNL NSAT

UNCLASSIFIED

UNCLASSIFIED

[redacted]

BNWL-840

UC-80, Reactor Technology
Special Distribution

HEAVY WATER REACTOR PROGRAM MONTHLY ACTIVITIES REPORT
MAY 1968

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

AEC Division of Reactor
Development and Technology Programs

NOTICE

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.

The information contained in this report has not received patent clearance by the U.S. Atomic Energy Commission or its contractors. However, this document is being made available to the Atomic Energy of Canada, Ltd., with the understanding that the information therein will be maintained in confidence by AECL and will be used only internally within AEC/AECL joint programs.

Although this report was prepared and distributed under Atomic Energy Commission Contract AT(45-1)-1830, it also includes related technical information submitted by other Atomic Energy Commission contractors.

By

The Staff of the Heavy Water
Reactor Program Office

H. Harty, Manager

June 30, 1968

PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON

[redacted]

UNCLASSIFIED

HEAVY WATER REACTOR PROGRAM MONTHLY ACTIVITIES REPORT

MAY 1968

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 under 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 HWRPO personnel.

PREVIOUS MONTHLY REPORTS

July and August 1967	BNWL-608
September 1967	BNWL-627
October 1967	BNWL-631
November 1967	BNWL-664
December 1967	BNWL-688
January 1968	BNWL-703
February 1968	BNWL-745
March 1968	BNWL-769
April 1968	BNWL-794

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

HEAVY WATER REACTOR PROGRAM MONTHLY ACTIVITIES REPORT
MAY 1968

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

H. Harty

I. SUMMARY

A. HEAVY WATER REACTOR PROGRAM ACTIVITIES

Program Management - 12000

The U.S. representative at the Halden Boiling Water Reactor (HBWR) reports that the first digital computer control of HBWR plant circuit has been successfully carried out. In the first test, the computer controlled the pressure in the feed water tank and responded properly to both normal and deliberate pressure perturbations. (12200)

General Studies - 13000

A program has been outlined for the analysis of safety problems of building a PHWR-PT in the U.S. It would involve evaluation of the CANDU-type of reactor with respect to U.S. regulatory requirements and safety criteria, and tentative proposal of criteria for HWR's in the U.S. (13100)

Testing of the XENCON code (for accuracy in predicting xenon oscillations) is continuing, and the source calculations and moderator and fuel temperature feedback mechanisms are functioning properly. Modifications are currently underway to stabilize the calculational model and accelerate source convergence. An investigation of the various parameters is being conducted in an attempt to decrease computing times. (13200)

The fuel cycle costs for UO_2 and metallic uranium fuels were calculated to determine the economic characteristics of these fuels and the incentive for investigation of metallic fuels. Comparison of the results shows that the optimized fuel cycle cost for metallic fuels is about 0.4 mills lower than for UO_2 fuels. The importance of optimizing a reactor concept can be seen. (13300)

Conversion of four Systems Analysis Task Force (SATF) HWOCR designs to HWPHW designs has been completed. Power costs were calculated for UO_2 fueled HWPHW designs. Power costs for uranium metal, ThO_2, and thorium metal fueled HWPHW reactors, although not based upon fully optimized fuel designs, are higher than for the UO_2 fuel case. (13300)

R&D Studies - 20000 thru 99000

The first "Thermos" assembly, a PNL-conceived specimen design for measuring in-reactor creep in biaxially stressed tubing, is being tested in an out-of-reactor corrosion test loop. Initial measurements indicate that the average diameter change of the tubular specimen from creep can be measured accurately by simple liquid displacement measurements, obviating the need for detailed and time-consuming dimensional measurements, except where desired for extended detail.

Work was initiated on the determination of flaw growth and fracture properties of pressure tubing. Fatique cracks were grown by cyclic internal pressurization in PRTR-sized tubes of Zircaloy-2 and aluminum. These crack growth data on pressure tubes were combined with crack growth data on analytically tractable sheet specimens and used to construct an initial thick-walled tube correlation of the absolute relationship between the stress intensity factor and crack length. These initial and preliminary results showed good agreement between the two materials and indicated this approach will allow flaw growth data on tubes to be supplemented substantially by results using simpler specimens. (21300)

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

Heat Transfer and Fluid Dynamics

In earlier experiments, the effect of rod spacing on the boiling burnout in rod bundle fuel elements was studied. Data were obtained for spacings of 0.080 and 0.040 in., pressures of 600, 1200, and 2000 psig, water mass velocities of 1, 2, and 3 x 10^6 lb/hr-ft^2, and outlet conditions ranging from 4 to 40%.

These data were analyzed to define the effects of rod spacing, pressure, mass velocity, and outlet quality on boiling burnout. The results showed that the boiling burnout heat flux was generally greater with 0.080 in. spacing than with the 0.040 in. spacing depending on pressure and mass velocity. This behavior is in agreement with the data of other investigators. However, some studies have indicated significant effects only for spacings less than 0.040 in. (TAC 3.1.5 - Effect of Rod Spacing on Boiling Burnout)

The computer code COBRA was developed earlier to enable the calculation of local enthalpy in rod bundle fuel elements and for the analysis and correlation of existing boiling burnout data. A comprehensive literature survey of boiling burnout investigations published since 1962 has provided a large amount of data which has been prepared for input into COBRA. The computer analysis of this data has started, and results should be available soon for comparison and correlation purposes. (TAC 3.1.6 - Subchannel Code Development and Local Subchannel Burnout)

Preliminary analysis of flow and enthalpy data for isolated subchannels of the BLW-250 geometry has been completed. Limitations in the flow measuring instrumentation, which resulted in scatter in the data, will be corrected for the final series of tests. (TAC 3.3.9 - Measurement of Flow and Enthalpy in the Subchannels of a BLW-250 Geometry Rod Bundle)

Two test sections to simulate the Pickering 28-rod bundle are being fabricated. Experimental operation with the 28-rod test assemblies is scheduled for the latter part of June. (TAC 3.3.12 - HWR 28 Rod Bundles)

Specifications for the motor generator sets to provide increased power capability have been written. It is expected that this equipment will be reconditioned and altered by late August. (Columbia University)

During May, the principal accomplishment was completion of programs for simulating operation of the Pickering reactor under two new control schemes. (TAC 4.3 - Analysis of Spatial Flux Control)

Measurements of the uniform temperature coefficient of a simulated burned-up UO_2 lattice at a pitch of 12.2 in. have been completed as well as foil irradiations at 23 and 80 °C (TAC 5.2 - Temperature Coefficients Measurements)

Eight tests were completed in a study of the effect of hydriding (~250 ppm) on the crack growth and fracture properties of Zr–2.5 wt% Nb pressure tubing. The cracks were caused to grow by cyclic internal pressurization. Comparing the results of these tests with earlier tests of unhydrided tubing showed that cold-worked Zr-Nb tubing retained substantial fracture toughness, (resistance to fracture in the presence of sharp fatigue cracks) even in the hydrided condition. The heat-treated tubing, on the other hand, exhibited severe reductions in fracture toughness when hydrided, allowing cracks less than 1/2 in. long to propagate at typical HWR pressure and hoop stress conditions. (TAC 6.9 - Evaluation of Zr–2.5 wt% Nb Pressure Tubing)

A uniaxial specimen of Zr–2.5 wt% Nb at 52.7 kg/mm^2 and 300 °C showed in-reactor creep rates 1/3 to 1/10 of those of out-of-reactor specimens. These results agree with earlier tests which indicate that the neutron flux decreases creep rates at high stresses compared to its accelerating effect on creep at low stresses. (TAC 6.10 - In-Reactor Measurements of Creep in Zr–2.5 wt% Nb Alloy)

Research is continuing to determine the dependence of swelling in U_3Si on temperature, hydrostatic pressure, burnup, burnup rate, and metallurgical state. Two controlled temperature-pressure swelling capsules designed to irradiate U_3Si in NaK at 625 and 725 °C at 50 psi were charged into a reactor. Fabrication is continuing on three additional capsules that will irradiate U_3Si at 1000 psi over the temperature range of 350 to 625 °C. Quantitative metallography is continuing on sections from three irradiated rods to determine the size-frequency relationship of the porosity to the total volume increase. (TAC 6.15 - U_3Si Irradiations)

Fabrication of the metallic fuel rod assemblies for irradiation in WR-1 is in progress. Sheath tubing has been fabricated, sectioned to length, annealed, and is in final inspection. Machining of the uranium pellet fuels is complete, and final inspection is being performed. Coextrusion of clad fuel with axial voids was successfully completed, and fuel sections are being examined to determine dimensional control, bond quality, and cladding uniformity. The fuel extrusions were cut to length, and beta heat treatment was completed. Electron beam welding conditions were established, and welding conditions for the TIG closure for the pellet fuels are being investigated. (TAC 6.16 - Uranium Alloy Fuel)

A literature review of particle transport in a condensing steam system was completed. A study of nucleation of water drops on particles and their subsequent growth has been initiated.

The preliminary theoretical investigation of condensation in steam-air systems progressed to a point where an experimental design is being considered. Interest in the boundary-layer equations describing condensation in laminar-flow systems has led to the application of analogue computers for their solution. (TAC 10.1 - Particle Transport to Surfaces by Condensing Steam)

II. HEAVY WATER REACTOR PROGRAM

A. PROGRAM MANAGEMENT

Foreign Technical Support - 12200
(Staff of HWRPO)

Halden, Norway

Reactor Operation. After Calibration of the newly loaded capsule rig IFA-117, the reactor was operated during the month at power levels up to 16 MW. By the end of April, the following integrated operation had been accumulated on the third fuel charge since the start of test fuel irradiation last November:

Operating time:	2215 hr
Integrated power:	908 MWd
Plant availability:	61%

The heavy water leak rate at the beginning of the month was 10 to 15 g/hr. After a temporary shutdown in the middle of the month, however, the leak rate increased to about 50 g/hr and was still increasing when the reactor was shut down. The leakages were traced to three valves in the subcooler circuit. The valves were repacked, and the leak rate was reduced to about 30 g/hr.

At the end of April the reactor was shut down as scheduled for general maintenance and loading of seven new test assemblies. Startup is scheduled on May 20.

The power level of individual test fuel assemblies during normal operation is usually measured with gamma thermometers. The relationship between the instrument signal and assembly power is determined during the initial in-pile calibration of the assemblies. Subcooled water is introduced into the channel to suppress boiling so that the gamma thermometer signal can be calibrated against assembly power as determined from channel flow and ΔT. Experience has shown, however, that the constants change significantly when the control rod configuration or core loading is altered. A method of recalibrating the gamma thermometers during normal operation has now been developed. Tests show that channel power under boiling conditions can be calculated quite accurately from the inlet and exit turbine signals and the inlet subcooling.

This method uses the known response of a turbine flowmeter to two-phase flow. If the slip ratio is known, the steam quality can be calculated from the comparison between the inlet and exit turbine speeds. Tests have shown that a constant slip ratio can be assumed over a wide range of operating conditions. Since all of the signals from an instrumented assembly are monitored by the process computer, this calculation can be done by the computer at regular intervals, and used to update the gamma thermometer calibration.

The first digital control of plant circuits was successfully carried out. In the first test, the computer controlled the pressure in the feedwater tank, in the same manner as the standard PID analogue controller normally used for this purpose. This loop was chosen because maloperation would not affect the safety of the plant. The digital control responded properly both to normal pressure fluctuations and to deliberate perturbations. The details of the control technique are described below. Based on the confidence gained from this test, four control loops were operated in DDC: reactor pressure, steam drum water level, steam generator water level, and the feedwater tank pressure. Experiments were performed in which steps in nuclear power and steam load were introduced with digital control of the loops and with conventional analogue control. The control quality was about the same in both cases; this was the desired result, since the digital control was designed to duplicate the functions of the analogue controller. The final objective of this work is to determine if digital control of a large number of loops might be cheaper than analogue control, particularly if a process computer is needed in any event for other functions such as optimization studies, etc.

<u>HWR Information Center - 12300</u>
<u>(B. B. Lane)</u>

Over 400 reports and journal articles have been abstracted and keypunched for key-word indexing. The 4000 cards are being proofread, and the detailed index should be off the computer within 10 days. Some subject headings for the input material have been added and others revised, based on the results of the previous run. A run of Canadian proprietary materials is also being prepared.

B. GENERAL STUDIES

General Safety Analysis - 13100
(W. R. Lewis)

To obtain an evaluation of the safety problems of building a PHWR-PT in the U.S. an analysis program has been outlined for implementation in FY 1969. The approach involves evaluation of a CANDU reactor type by a U.S. firm familiar with U.S. regulatory requirements. Canadian safety practices would be reviewed and compared with U.S. criteria for LWR's. Tentative U.S. criteria for HWR's would be proposed and a safety analysis of a typical CANDU reactor type prepared in a manner similar to safety analyses for U.S. LWR plants.

Physics - 13200
(W. C. Wolkenhauer)

Boiling H_2O Reactivity Coefficients*

Material buckling measurements have been performed using four coolant materials to simulate boiling light water coolant. The fuel assembly consisted of 31 rod clusters of a burned-up fuel mockup with Pu/U weight fraction of 0.00259.

The measurements were made with a lattice of 19 fuel assemblies in the SE on a 12.12 in. triangular pitch. Measured values of the square of the reciprocal of the vertical thermal neutron relaxation length, H^2, are reported in Table I. The moderator purity corresponding to each measurement is also given. Effective radial bucklings for the SE have been calculated using the PDQ-05 computer code. Input parameters for PDQ-05 were obtained from cell calculations performed using the HAMMER code. Table I also contains material buckling values determined from the measured values of H^2 and PDQ calculated values of $B_R{}^2$.

---

* *Work Performed at Savannah River Laboratory*

TABLE I. Buckling Measurements for Various Coolants in SE Lattice on 12.12 in. Triangular Pitch

Coolant	Moderator Purity, mole % D_2O	Measured H^2, μb	Radial[a] Buckling B_R^2, μb	Measured B_m^2, μb
Air	99.62	573	903	330
H_2O	99.62	886	923	37
39.4% H_2O[b]	99.61	757	918	161
D_2O	99.61	644	912	268
D_2O	99.34	668	912	244

(a) *Effective radial buckling calculated using PDQ-05*
(b) *39.4 mole % H_2O - 60.6 mole % D_2O mixture*

Optimum Spatial Flux Control

Description and Developmental Progress of XENCON.* Testing of the XENCON program with the original sample problem is continuing. The latest results obtained with the code show that the source iteration calculations are being performed correctly, and that the fuel temperature and moderator temperature feedback mechanisms have been included properly. These results have been verified by comparisons with hand calculated numbers. However, these calculational results have pointed out two problem areas, the first of which was anticipated beforehand because of prior operational experience with FLARE type calculations. A printout of the source distribution after each iteration revealed that after a certain number of iterations (approximately 15 to 20), the calculations exhibit an oscillatory behavior, with the values in each node varying by about 1% between successive iterations. This type of result is inherent in FLARE type calculations. Hand calculations indicate that if average nodal source values from two successive iterations are used as the source guess for the subsequent iteration once these calculational oscillations have

* *Work performed by Combustion Engineering*

set in, the oscillations will disappear very rapidly. This averaging procedure should result in accelerated source convergence, and it has been programmed into XENCON, but no results have yet been obtained.

The second problem area deals with the time required by XENCON to perform a source calculation. Typical times encountered so far were found to be approximately 10 sec per iteration. It appears at present that this is the maximum time required for this calculation for a full 24 x .4 x 10 node description. The main problem associated with this large running time is due to the large number of tape units that must be manipulated during the calculations, which is a result of the number of variables that enter into the calculational scheme. In an attempt to reduce running times, a new type of procedure for tape reading, which is available in FORTRAN programming, is being investigated. At the same time, other approaches for reducing machine time are being investigated. These include reducing the maximum number of nodes in the XY plane from the present (24 x 24). Also, the size of the three-dimensional matrices of some of the variable parameters used in the calculational routines can be eliminated or reduced by the introducing of valid assumptions or approximations. These cutbacks would allow for a reduction in the amount of information that is presently stored on tape units and would result in a more economic use of fast core memory.

A new test case for XENCON has been set up, and the corresponding INCON tape has been generated. This case is much more realistic from the standpoint of actual application of the program than the original sample case, and it will be used through the remainder of the debugging and check-out phase after the code becomes operational again.

Node Size Study. The two-dimensional node size study as described in previous reports is continuing. The appropriate FLARE calculations have been set up for the three different node sizes, and a trial-and-error technique is currently being used to determine the leakage parameters for each node size for each of the two FLARE type transport kernels. In addition to this, another multigroup diffusion theory calculation has been made with the same core composition as the original problem but with a different control rod configuration in order to give a much higher peak-to-average power distribution. The purpose of these calculations is the same as that reported in the last report for the one-dimensional study in which a set of leakage parameters was selected for the most peaked distribution. These parameters were used for flatter distributions to determine if satisfactory results would still be obtained. The trial-and-error method for selecting the leakage parameters for the two-dimensional distribution with the highest peak has begun.

Economic and Initial Design Studies - 13300

Fuel Cycle Costs
(J. R. Young)

The fuel cycle costs for UO_2 and metallic uranium fuels were calculated to determine the economic characteristics of the fuels and the incentive for investigation of metallic fuels for HWR's. The fuel element descriptions and the characteristics of the irradiated fuel were obtained from document AI-CE-MEMO-59. The fuel cycle economic data (unit costs of fabrication, shipping, and reprocessing) were obtained from the economic correlations of the Advanced Converter Task Force. The specific economic data are shown in Table II, and the results of the analysis appear in Figure 1.

TABLE II. Economic Data for Fuel Cycle Analysis

	Metallic Uranium Fuels	UO_2 Fuels
Unit cost of fuel fabrication, $/kg	14.00	36.00
Unit cost of fuel reprocessing, $/kg	11.00	15.50
Unit cost of fuel shipping, $/kg	1.10	2.30
Reactor plant factor	0.80	0.80
Reactor thermal efficiency, %	30.00	30.00
Interest rate on feed material, %	16.00	16.00
Interest rate on working capital, %	12.80	12.80
Average heat generation rate, MW/tonne	16.50	33.00
Market value of fissile Pu, $/g	5,10,15	5,10,15
Unit cost of separative work, $/unit	26.00	26.00

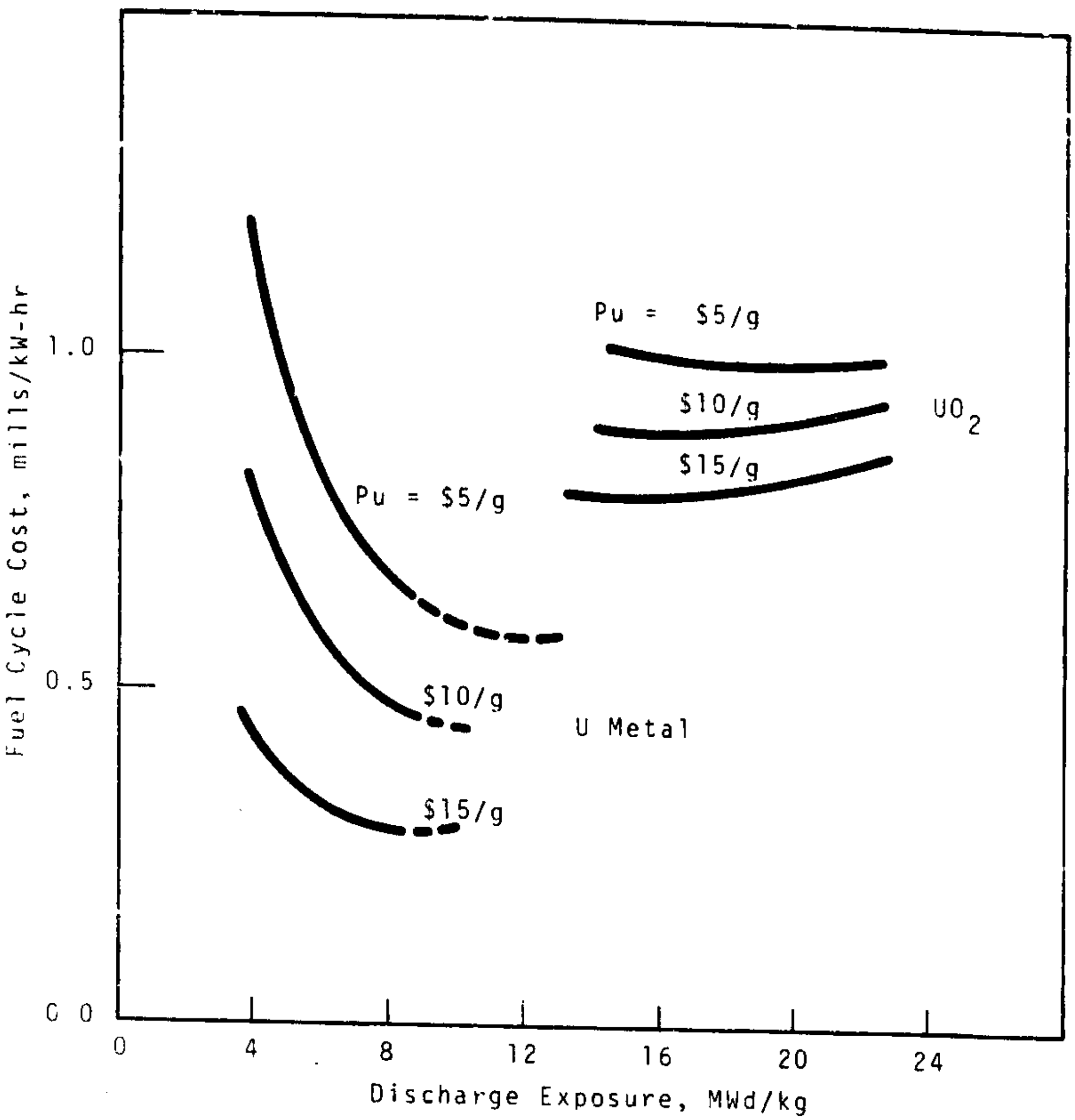


FIGURE 1. Fuel Cycle Costs, UO_2 and Metallic Uranium Fuels

In general, insufficient data were available to permit calculation of the optimum exposures for metallic uranium, but extrapolation of the results (the dashed lines) indicates that the fuel cycle costs for metallic uranium at an optimum exposure of about 10 MWd/kg are 0.4 to 0.5 mills/kW-hr lower than for UO_2. These fuel cycle costs do not include the heavy water inventory costs.

HWOCR Conversions*
(G. A. Last)

Last month, power costs were presented comparing a UO_2-fueled HWPHW design with a UC-fueled HWOCR design and a PWR design. Comparisons were shown for base-load and peak-load-operation, constant and declining fuel cycle costs, and for 1976 and 1980 startup.

The required calculations have now been completed for converting all of the Systems Analysis Task Force (SATF) HWOCR designs to HWPHW designs, and power cost comparisons between these various designs are presented here. Costs are for base-load-operation, declining costs as calculated with the FUELCO code, and a 1976 startup. The power cost calculations were made using the PACTOLUS code.

The basic assumption for this study was that the lifetime material balance data prepared for the HWOCR designs could be used to define the fuel requirements for comparable HWPHW designs because of similarities in nuclear properties between the two design concepts (substitution of heavy water for the organic coolant would be approximately compensated for by substitution of Zircaloy pressure tubes for SAP pressure tubes). However, fuel throughput and inventory adjustments were required because of different thermal efficiencies and specific powers.

Thermal-hydraulic limited design parameters were defined for the four HWPHW systems of interest based on the HWOCR fuel designs. Estimates of capital cost adjustments for converting

* *Work performed by Pacific Northwest Laboratory Mathematics Department*

the HWOCR designs to HWPHW designs were made. Both the UO_2 and ThO_2 fuels were constrained to a linear power rating of 18 kW/ft. Both the U-metal and Th-metal fuels were found to be DNB (departure from nucleate boiling) limited.

The design features of the four HWPHW designs that evolved from this study are summarized in Table III. The designs are not necessarily optimum, but are based on optimized HWOCR designs. Because of similarities in the two systems it is believed that these HWPHW designs are close enough to optimum that it is unlikely that large cost differences would evolve from a more thorough optimization study. The greatest extrapolation involved conversion of the UC-fueled HWOCR to a UO_2-fueled HWPHW. Other fuels were the same in both systems.

TABLE III. 1000 MWe HWPHW Design Features

Fuel Material	UO_2	ThO_2	U-Metal	Th-Metal
Fuel Description	37-rod	37-rod	3-ring annular	4-ring annular
Capital Cost, \$/kWe	132	135.7	145.9	130.8
Thermal Power, MW	3280	3280	3280	3280
Thermal Efficiency, %	30.5	30.5	30.5	30.5
Core Inventory, MW of U or Th	100	90	262	122.5
Specific Power, MW/tonne	32.8	36.4	12.5	26.8
Number of Pressure Tubes	596	596	653	598
D_2O Inventory, tonne	355	402	587	446
Core Inlet Pressure, psi	1780	1780	1780	1780
Core Pressure Drop, psi	98	98	193	80
Core Inlet Temperature, °F	510	510	510	510
Core Outlet Temperature, °F	577	577	577	577
Coolant Flow Rate, lb/hr	1.31×10^8	1.31×10^8	1.31×10^8	1.31×10^8
Maximum Heat Flux, Btu/fr-ft^2	4.22×10^5	4.22×10^5	5.05×10^5	4.4×10^5
Maximum Fuel Temp Temperature, °F	(designed to 18 kW/ft)		925	780

For calculation of power costs the SATF ground rules were used. The basic cost assumptions and estimates are shown in Table IV. For comparative purposes the SATF 1970 PWR design is included.

TABLE IV. Basic Cost Data

U_3O_8 Cost, $/lb	8.00
Separative Duty, $/kg	26.00
Pu Value, $/g	10.00
D_2O Cost, $/lb	17.50
Fixed Charge Rates	
Depreciable, %/year	13.6
Non-depreciable, %/year	12.6

	PWR	HWPHW UO_2	HWPHW ThO_2	HWPHW U-Metal	HWPHW Th-Metal
Capital Cost, $/kWe	132.7	132.0	135.7	145.9	130.8
Fabrication Cost,[a] $/kg					
Initial	55.0	51.7	69.6	33.7	57.9
Final (30 year)	37.0	27.7	36.1	10.3	26.3
Reprocessing,[a] $/kg					
Initial	27.2	24.6	48.7	23.6	48.4
Final (30 year)	13.7	11.7	21.2	10.3	21.2

(a) *[illegible] Startup basis*

Fuel mass-balance data was available at several fuel exposures for the UO_2 (UC originally), ThO_2, and Th-metal designs but only for one exposure for the U-metal design based on natural uranium. The power generating costs as calculated with PACTOLUS are plotted as a function of fuel exposure for the UO_2, ThO_2, and Th-metal designs in Figures 2, 3, and 4 respectively.

Additional fuel mass-balance data at two higher exposures were calculated for the HWPHW UO_2 fuel design with the BNW ALTHAEA program. Power costs calculated with this data are also plotted in Figure 2 and mesh very well with HWOCR-UC based data. This data confirms the validity of using the HWOCR data for HWPHW fuel cost calculations as well as demonstrating optimum

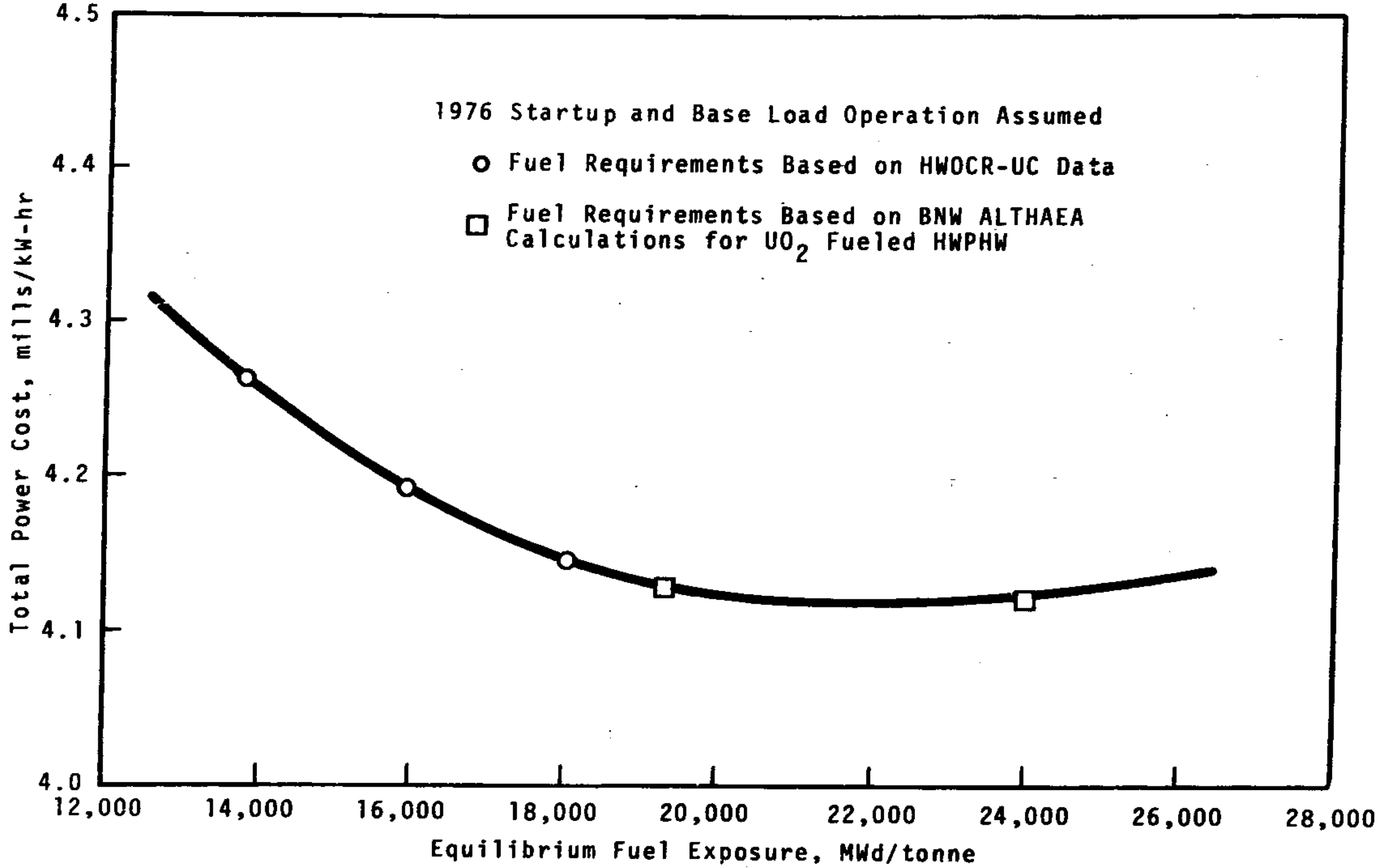
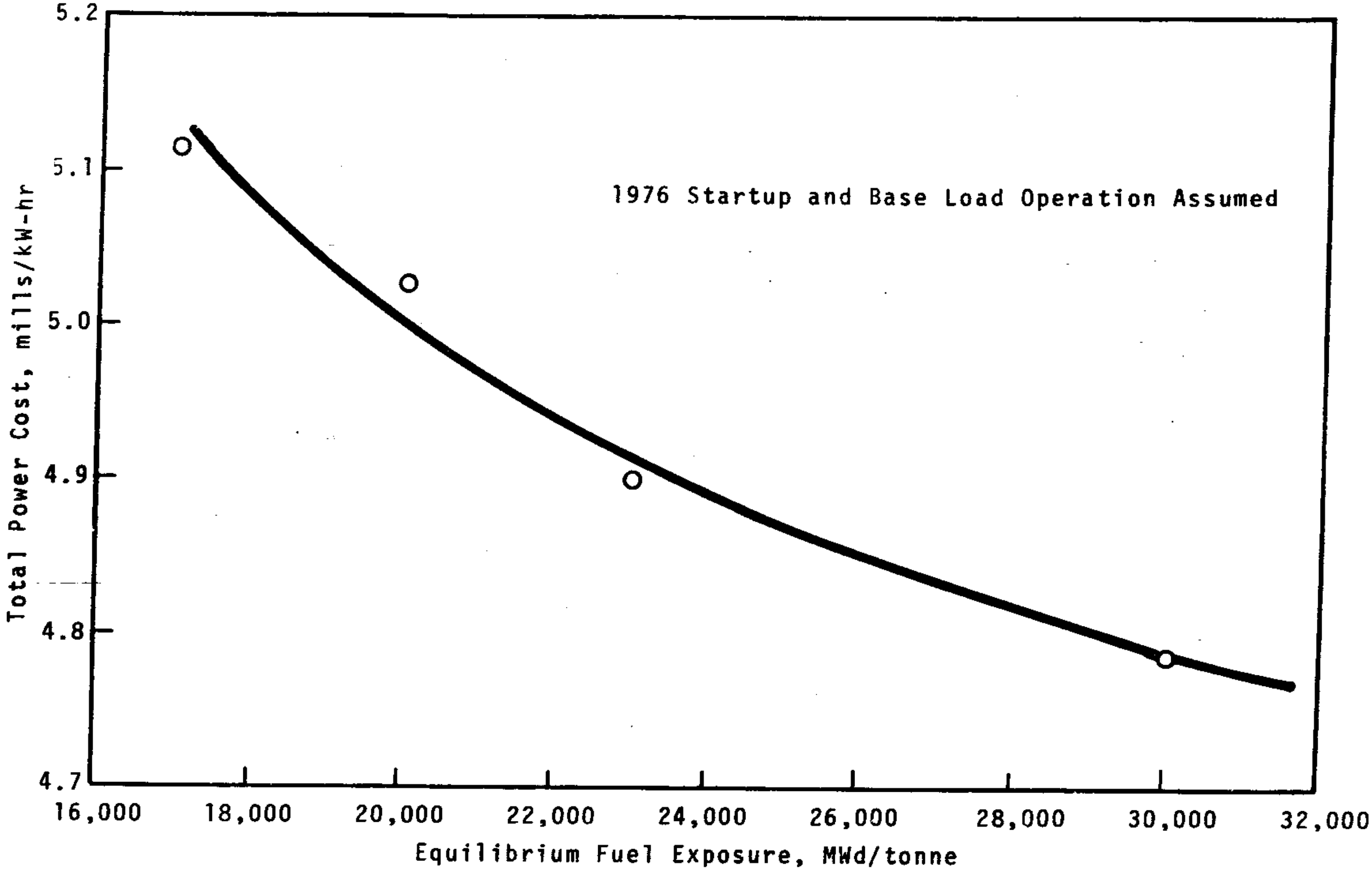


FIGURE 2. Power Cost for UO_2 Fueled HWPHW Reactor (Based on HWOCR-UC design conversion)



<u>FIGURE 3</u>. Power Cost for ThO_2 Fueled HWPHW Reactor (Based on HWOCR design conversion)

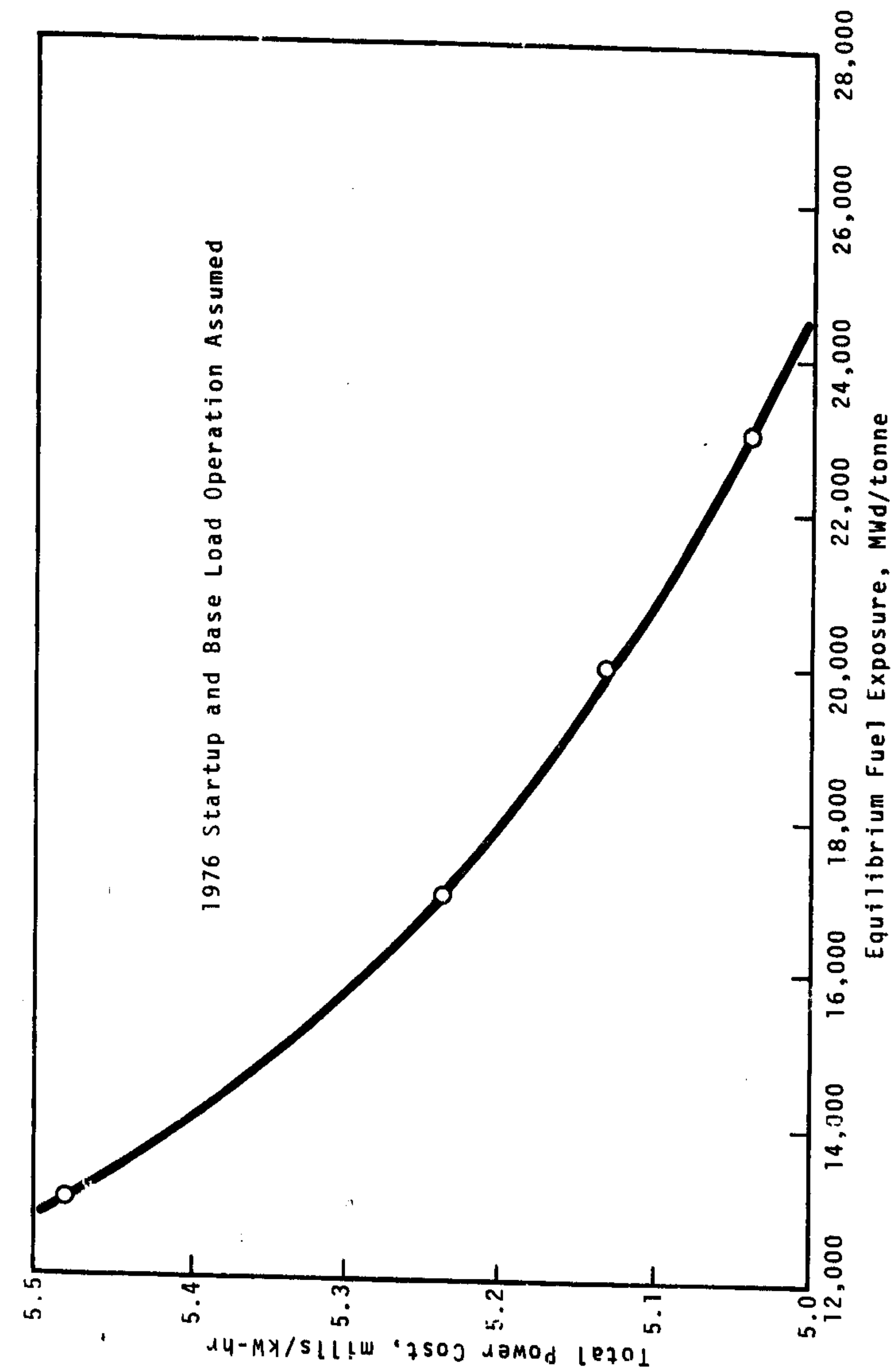


FIGURE 4. Power Cost for Th-Metal Fueled HWPHW Reactor (Based on HWOCR design conversion)

exposure for the UO_2 fuel to be approximately 22,000 MWd/tonne. The costs are a little lower than those reported last month for this design as the result of a correction in the D_2O inventory requirement.

Figures 3 and 4 show that the ThO_2 and Th-metal fuels had not reached theoretical optimum at the maximum exposure calculation.

Power costs for the four HWPHW designs are compared with PWR costs in Table V. The comparison is based on optimum exposure for the UO_2 fuel and the highest, but not necessarily optimum, exposure calculation available for the others.

TABLE V. Reactor Power Cost Comparisons for 1976 Startup

Base load operation and
declining fuel cycle costs assumed

	PWR	HWPHW UO_2	HWPHW ThO_2	HWPHW U Metal (Natural)	HWPHW Th-Metal
Equilibrium Fuel Exposure, MWd/tonne	30,000 declining to 20,000	22,000	30,000	5,200	23,000
Power Cost, mills/kW hr					
Depreciable Plant	2.57	2.56	2.63	2.82	2.53
Operating and Maintenance	0.29	0.38	0.38	0.39	0.37
Fuel Cycle Including D_2O	1.47	1.18	1.78	1.27	2.14
Total	4.34	4.12	4.79	4.48	5.04

C. RESEARCH AND DEVELOPMENT STUDIES - 20000 THRU 99000

Pressure Tubes - 21300
(R. P. Marshall)

Creep of Pressure Tubing*

In this task, thick-walled tubular specimens will be internally pressurized to measure various creep and stress-rupture

* [illegible]

parameters, such as rate and magnitude of primary and secondary creep, time and strain to tertiary creep and to rupture, and the sensitivities of these properties to stress, stress system, temperature, neutron flux, neutron fluence, and material factors. A prime goal of this task is to define how creep to some strain limit or to rupture will affect pressure tube lifetime, and to seek ways to maximize reliable pressure tube life as controlled by creep. This includes, therefore, the effects of material and environmental parameters on mean creep behavior and the dispersion in behavior expected in a large sample, i.e., the many tubes in a large HWR.

An experimental test assembly (Thermos specimen) designed for obtaining in-re ctor creep data from biaxially stressed tubing is under development. This assembly consists of two concentric tubes separated by an evacuated annular space between the inside of the outer tube and the outside of the inner tube; the ends are sealed by welding or brazing spacer rings at the ends of the tubes. The wall thickness of the inner tube is selected so that the environmental pressure on the inside surface and the vacuum on the outside surface produces stresses large enough to cause creep. The outer tube has a wall thick enough so that there is no creep under the action of hydrostatic pressure on the outside surface. This assembly has the substantial advantage that creep-to-rupture tests can be run in laboratory loops or reactor channels without the release of damaging amounts of stored energy to the test environment, including a "driver" fuel element which can be located within the inner tube during reactor tests. With appropriate "driver" fuel design, high fast neutron exposures can be achieved.

The outer tube, which serves as a "load frame" for the inner tube creep specimen, may, subsequent to a creep test, be used for postirradiation crack propagation tests. Stress riser slots or simulated defects can be placed in the outer tube surfaces, provided that they do not compromise the annular vacuum space between the inner and outer tubes. Here, again, failure of the [illegible] cause damage to the environment.

Creep deformation measurements may be derived for this type of assembly by measuring changes in the liquid displacement of the assembly. Micrometer measurements of the inside diameter of the inner tube may also be used, but such measurements are often difficult to obtain with good accuracy after the assembly has been irradiated. Changes in displacement at periodic intervals can be measured by weight measurements in air and in water. If corrections can be made for corrosion weight gains, crud deposition, etc., subsequent measurements need only be made in water. In-reactor corrosion testing efforts are expected to provide the necessary corrections.

An experimental "Thermos" assembly has been fabricated and is being tested in a corrosion test loop. The loop operates at a pressure of about 1100 psig and temperatures corresponding to the ETR-G7 loop for each reactor cycle. At this pressure the hoop stress on the creep specimen of the Thermos assembly is about 27,500 psi. The initial run was about 70 hr at 272 °C. After this run the average (micrometer) measured change in diameter of the inner tubular creep specimen was 6.0 mils. The average change in diameter calculated from "displaced" weight changes was 4.9 mils. During three additional exposures the average diameter changes were 11.8, 4.0, and 2.3 mils by micrometer measurements compared to 11.6, 5.0, and 2.2 mils by displacement measurements, respectively. A fifth cycle will be completed in early June. So far the cumulative change in diameter is 24.1 mils by dimensional measurements as compared to 23.7 mils from "displacement" weight changes. These quantities are in rather good agreement and suggest that indeed "displacement" weighing can be satisfactorily used for creep measurements in the Thermos assembly.

<u>Flaw and Fracture Evaluation of Pressure Tubing*</u>

This task will develop information or factors controlling flaw growth, the rates of flaw growth, and the effects of

* [illegible] Engineering Materials and Mechanics [illegible]

naturally grown flaws on the fracture behavior of Zircaloy pressure tubing. Internally pressurized pressure tube sections and pressure tube coupons will be used to study flaw geometry, material, thermal, neutron flux, neutron fluence, and Chemical stress cycle effects. Comparison of the crack growth rates and stress intensification factors will establish feasibility of supplementing pressure tube data with the more efficient coupon tests.

Work has been initiated on a technique that should allow the prediction of crack propagation behavior in a pressure tube based on the results of simple flat test specimens. Using this technique, it should be possible to predict the behavior of in-reactor pressure tubes undergoing irradiation as well as unirradiated tubes.

At the present time, there is no satisfactory solution for the stress intensity factor (K) for a tubular configuration, and it is therefore not possible to treat the crack propagation behavior of tubes analytically. Several attempts have been made, both analytically and empirically, but to date, the results have not been universally applicable to a wide range of tube sizes and materials. Some of the initial work in this program will attempt to correlate the cracking behavior of analytically tractable configurations (flat sheets etc.) with the equivalent behavior in a pressure tube. This correlation would be used to supplement flaw growth tests on pressure tubing with tests of flat sheet specimens that are more efficient in testing material and environmental parameters such as texture, hydride orientation, and presence of neutron flux and neutron fluence. The technique should be applicable to all geometries of circular tubing and to many materials.

Some preliminary analytical work has been done on this method, and it will be described briefly. It should be emphasized that the results presented are based on insufficient data and are for illustrative purposes only. Experimental fatigue crack propagation data were measured for three PRTR-sized tubes of aluminum and Zircaloy-2 (Figure 5). These data were analyzed on the basis of recent literature results that show that the instantaneous crack growth rate in any material and specimen geometry during cyclic loading is controlled by the fluctuations in the stress intensity factor (ΔK) around the crack border. This means that when similar crack growth rates are observed in different geometries, the ΔK must be the same regardless of the experimental conditions of specimen geometry, etc. Thus, the crack growth rates observed in the pressure tubes (instantaneous slopes of curves in Figure 5) were related to ΔK by using a "material calibration" of crack growth rate against ΔK from analytically tractable specimens (Figures 6a & b). This derived ΔK value and the crack lengths associated with each data point were then used to construct the basic tubular calibration of ΔK versus crack length, Figure 7. While Figures 6a & b are material calibrations and independent of specimen geometry, Figure 7 is a specimen calibration and should be independent of material. It is believed that the data for Zircaloy-2 and 6061-T6511 aluminum (Figure 7) show some deviation between the two materials because the material crack propagation properties (Figures 6a & b) have not been determined accurately yet for either of the materials, i.e., observe the scatter in Figure 6b.

Once the empirical solution for ΔK in thick-walled tubes has been accurately derived and tested, it would become the means of converting crack growth versus crack length data on sheet specimens (like Figures 5, 6a & b) under new environmental and/or material conditions to the equivalent crack growth rate versus crack length for tubes under these same conditions.

HWRP Fabrication Specification for Zircaloy Pressure Tubing*

An HWR pressure tube fabrication specification is being prepared for the fabrication of high quality Zircaloy pressure tubing for HWR service. The specification is being based primarily on

---

* *Work performed by PNL Engineering Materials and Mechanics Section*

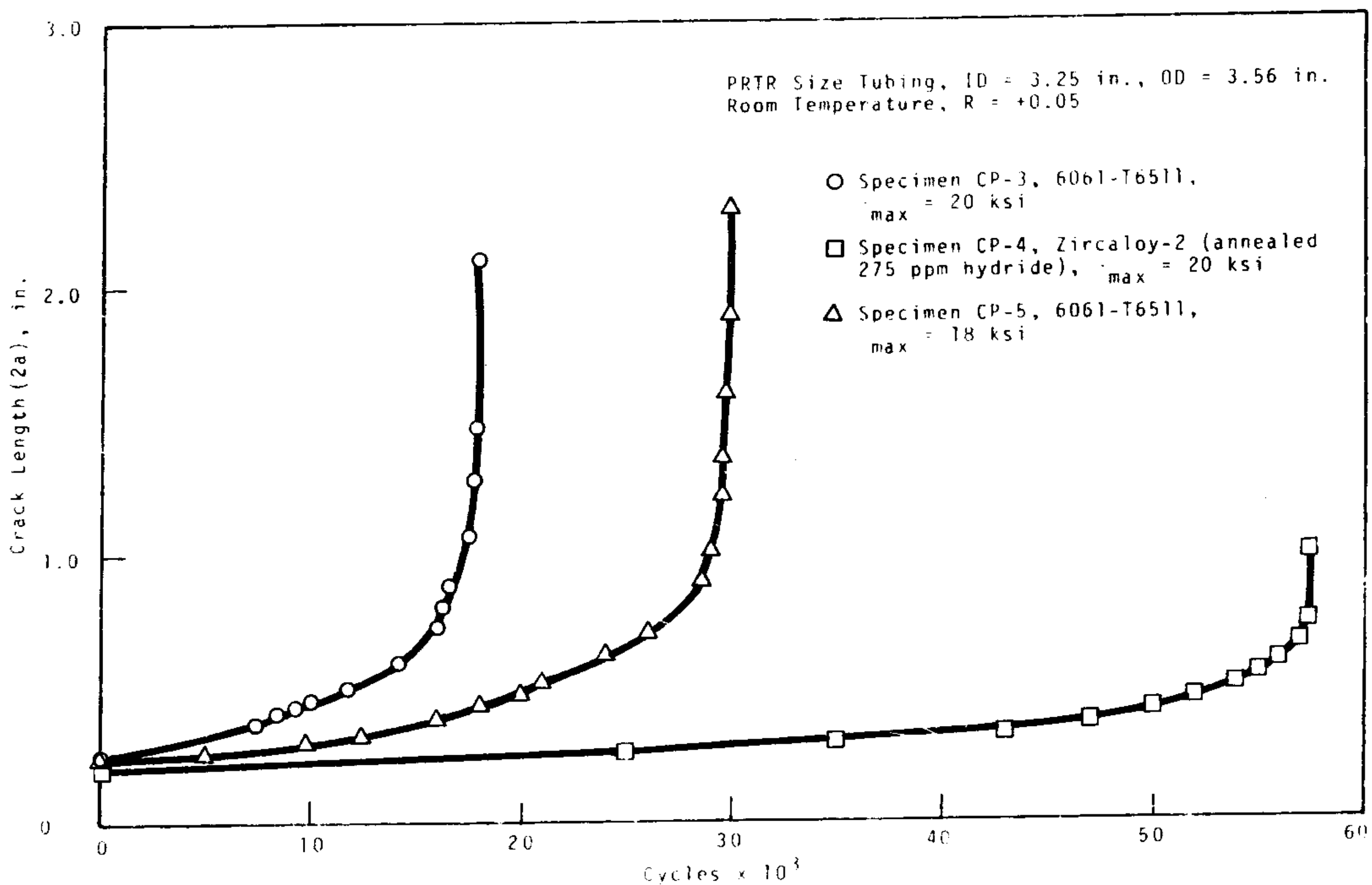


FIGURE 5. Fatigue Crack Growth Data for Thick Walled Tubes

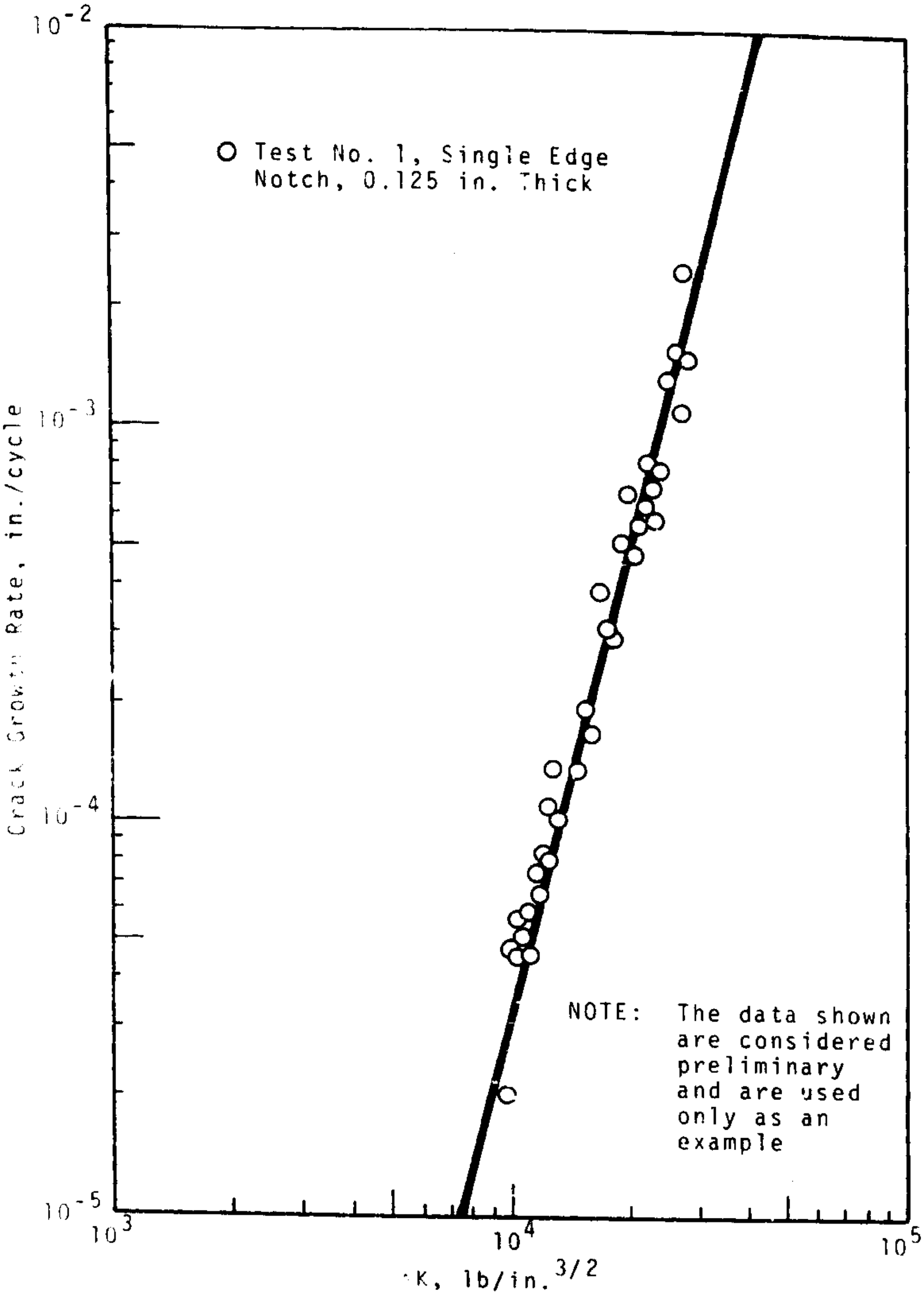


FIGURE C1. Crack Growth in a Sheet Specimen of Aluminum

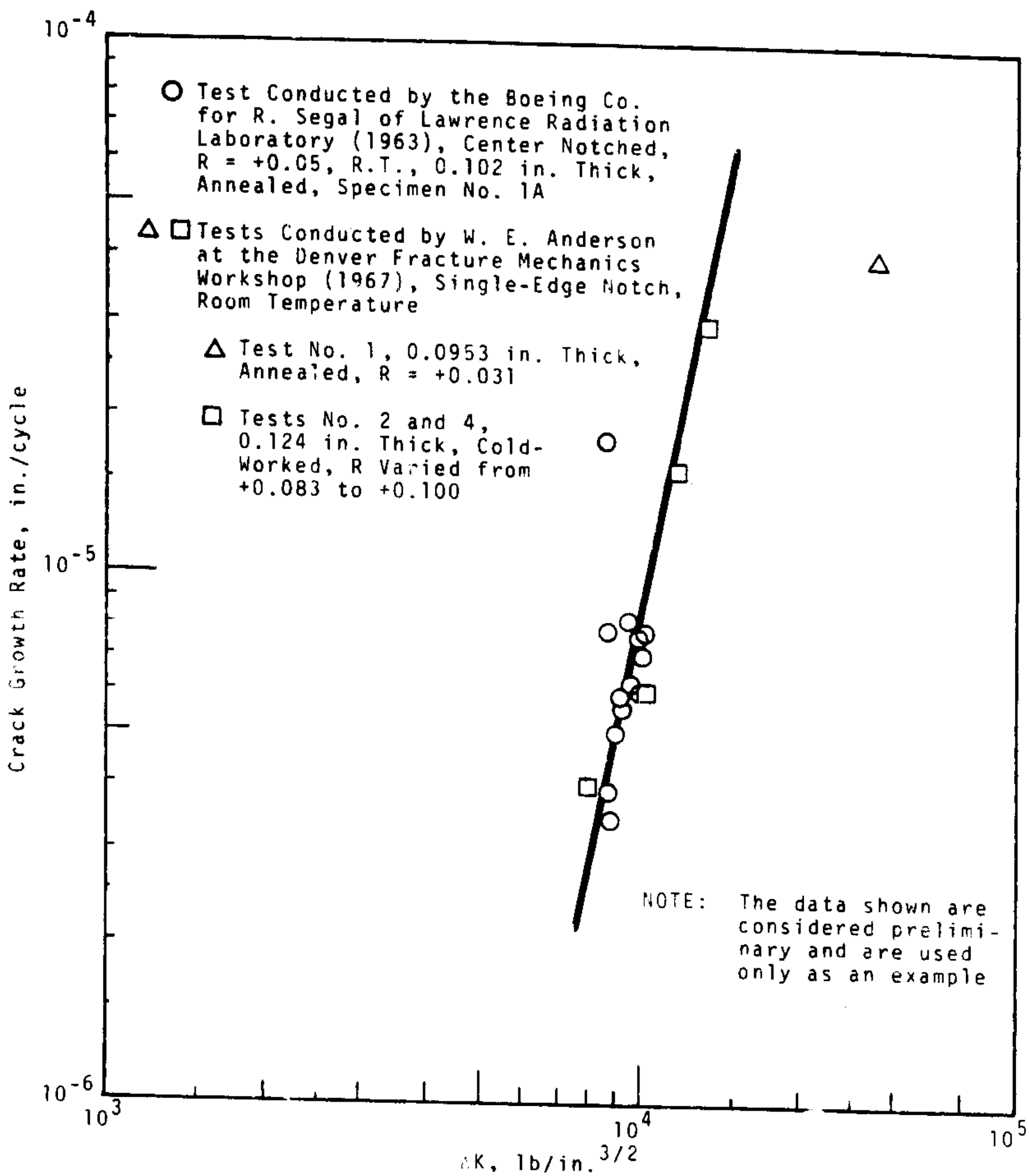


FIGURE 6b. Crack Growth in Sheet Specimens of Zircaloy-2

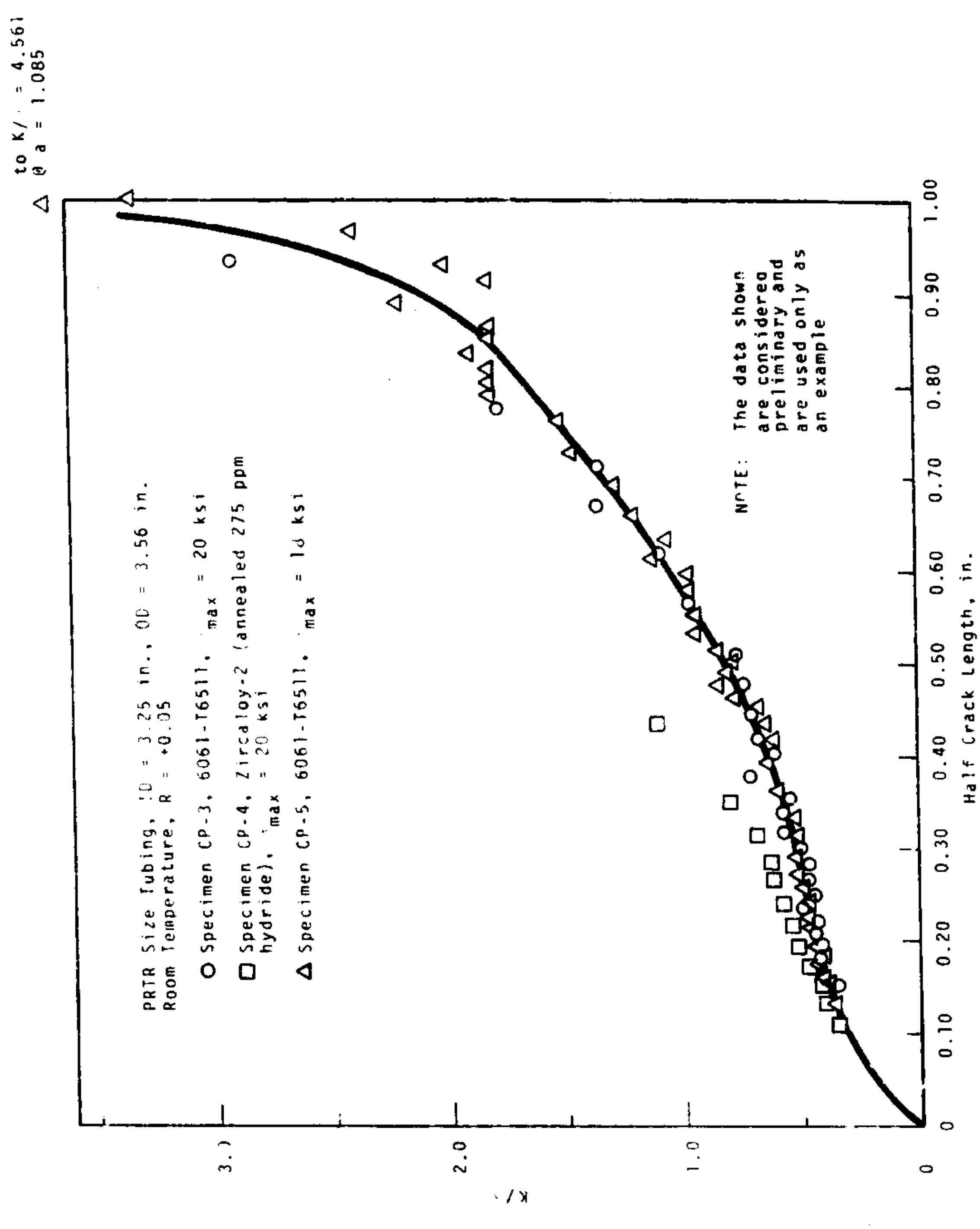


FIGURE 7. *Stress Intensity Factor for PRTR Size Tubing*

the compilation and review of the specifications used for Pickering, PRTR, and N-Reactor pressure tubes, and the appropriate ASTM specifications (reported in HWRP Monthly Report BNWL-703). There will be two major sections to the specifications. The first will contain those items necessary to assure high quality pressure tubing where the required limits for the specification have been defined. The second section will contain items that are candidates for the first section but cannot be included because their effect on tube life or reliability has not been defined, or the technolog required to formulate a specification is not available in a suita form for commercial use.

<u>Main Heat Transfer System - 31000</u>
<u>(W. R. Lewis)</u>

A revised task was prepared for the FY 1969 HWRP on the mai heat transfer system. This activity provides for the development of technology, systems, and components needed to upgrade currently acceptable light water reactor main heat transfer systems and/or components in those areas of special importance for HWR's. Emphasis will be placed on problems that are unique to HWR large systems such as those relating to detection and control of heavy water leakage, performance, and reliability. Problem areas that are common to both heavy water and light water reactors will, in general, be excluded except for maintaining cognizance of LWR developments. For large HWRDPR's, the required performance, reliability, and size of components and systems for dual purpose applications will be developed. In subsequent years a continued emphasis will be on collecting and analyzing equipment performance data with an increasing experimental program on leakage control.

Work continued during the month on status of technology for seals and leakage control methods.

III. USAEC/AECL COOPERATIVE PROGRAM ON HEAVY WATER REACTORS

A. HEAT TRANSFER AND FLUID DYNAMICS

Effect of Rod Spacing on Boiling Burnout - T.C 3.1.5

Pacific Northwest Laboratory
(D. R. Dickinson)

The purpose of this program is to determine the effect of the spacing between rods on the burnout heat flux in heat transfer from rod-bundle nuclear fuel elements to water. The analysis of the data obtained in the tests reported earlier has been completed, and a summary of these tests and significant results are presented below.

The test assembly used in these experiments consisted of an electrically heated inner rod surrounded by a multilobed outer wall which simulated the four adjacent fuel rods and four adjacent flow channels in a multirod fuel bundle with square pitch. Interchangeable center rods with diameters of 0.504 and 0.584 in. were used to produce spacings of 0.080 and 0.040 in., respectively. The equivalent diameters of the flow channel were 0.340 and 0.276 in., and the heated length was 60 in. Spacing between the inner rod and the outer wall were maintained by ceramic spacers attached to the inner rod at 12 in. intervals. The test section was designed so that boiling burnout would occur preferentially on the inner rod. The heat flux on the four inner lobes of the outer wall (facing the rod across the gap) was 86% of that on the inner rod, and the heat flux on the four outer lobes (facing the main flow channels) was 83% of that on the inner rod.

At each spacing, boiling burnout heat fluxes were determined at pressures of 600, 1200, and 2000 psig mass velocities of 1, 2, and 3×10^6 lb/hr-ft^2, and inlet conditions from 300 Btu/lb subcooled to 32% quality. The outlet quality at burnout was in the range of 4 to 40%; the burnout heat flux was 0.12 to 1.08×10^6 Btu/hr-ft^2.

The results of these experiments are summarized as follows:

- Effect of Rod Spacing

  The burnout heat flux was generally greater with the 0.080 in. than with the 0.040 in. spacing. The magnitude of the difference depended on pressure and flow. At 1200 psi, 1×10^6 lb/hr-ft^2, the burnout heat flux was about 10% higher with the wider spacing; at $2 \times 3 \times 10^6$ lb/hr-ft^2, the burnout heat flux was 30 and 40% higher, respectively. At 2000 psi and 2×10^6 lb/hr-ft^2, the observed effect of spacing on burnout heat flux was negligible. No conclusions could be obtained from the 600 psi data because of data scattering. The increase in burnout heat flux with increased spacing is in agreement with the data of other investigators. However, some of the other studies have shown a significant effect only for spacings less than 0.040 in.

- Effect of Pressure

  The burnout heat flux increased greatly with decreasing pressure at constant quality. At a mass velocity of 2×10^6 lb/hr-ft^2, the burnout heat flux was 50% higher at 1200 than at 2000 psig for the 0.080 in. spacing; 25% higher for the 0.040 in. spacing. The burnout heat flux was also higher at 600 than at 1200 psig, but the data at 600 psig were too scattered to assign a quantitative value.

- Effect of Mass Velocity

  At a constant outlet quality, the burnout heat flux as a function of mass velocity passed through a minimum at a mass velocity of about 2×10^6 lb/hr-ft^2 with the 0.080 in. spacing; the burnout heat flux at 1×10^6 and at 3×10^6 was 10% higher. When the spacing was reduced to 0.040 in., the minimum was shifted to a higher mass velocity. The burnout heat fluxes were nearly the same at 2×10^6 and 3×10^6 lb/hr-ft^2, but 30% higher at 1×10^6 lb/hr-ft^2.

- Effect of Outlet Quality

  The burnout heat flux decreased with increasing quality. (The few exceptions to this are probably erratic points.)

The change in burnout heat flux with rod spacing did not depend significantly on quality.

- Effect of Unheated Surface

  Two runs were made at the 0.080 in. spacing with the outer wall of the test section unheated. The burnout heat flux was 10% higher than with both surfaces heated. This is at variance with other burnout data, in annular geometry, which show a lower burnout heat flux with one surface unheated.

All the burnouts with the 0.040 in. spacing and most of those with 0.080 in. spacing occurred at the downstream end of the inner rod at points adjacent to one or more of the gaps. In some of the runs at 0.080 in. spacing (i.e., at 1200 psig, 3×10^6 lb/hr-ft^2 and at 2000 psig, 2×10^6 lb/hr-ft^2) burnout occurred on the outer lobes of the outer wall or occurred on both surfaces simultaneously. In one test (with only the inner rod heated), burnout occurred 9 in. upstream from the end of the rod.

Burnout appeared first as intermittent minor excursions in the surface temperature. When the heat flux was increased further, these excursions became larger and closer together giving a sawtooth graph of wall temperature versus time with a frequency of about 1 cycle/sec. Still further increases in heat flux resulted in fast temperature rises which could be arrested only by reducing power. The burnout heat flux reported here is that at the first appearance of these intermittent temperature excursions.

The test section is being rebuilt to replace the outer wall which was damaged by arcing at the end of these experiments. Following the replacement of this component, the test section will be used to measure burnout heat fluxes with a spacing of 0.015 in. for comparison with the data obtained at 0.040 and 0.080 in.

Subchannel Code Development and Local Subchannel Burnout - TAC 3.1.6
Pacific Northwest Laboratory
(A. S. Neuls)

The computer code COBRA was developed at PNL in FY 1967 to enable the analysis of local coolant conditions in rod bundle fuel elements by considering the effects of turbulent mixing and boiling diversion cross-flow between coolant subchannels. 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. During the current fiscal year, this code was modified, improved, and implemented in preparation for applying it to various selected rod bundle boiling burnout data.

A comprehensive literature survey of the available boiling burnout investigations of rod bundles and annular configurations published since 1962 has provided a large amount of data. Preparation of computer input containing the necessary geometric and operating factors from published investigations considering rod bundle fuel elements has been completed, and computer analysis of the data has started. However, preliminary use of the code revealed some minor program problems which have since been corrected.

The COBRA calculations of local conditions should proceed rapidly, and results should be available soon for comparison and correlation purposes.

Critical Heat Flux in a 9-ft Bundle - TAC 3.3.8
Columbia University Engineering Research Laboratory
(J. E. Casterline)

No report was received this month.

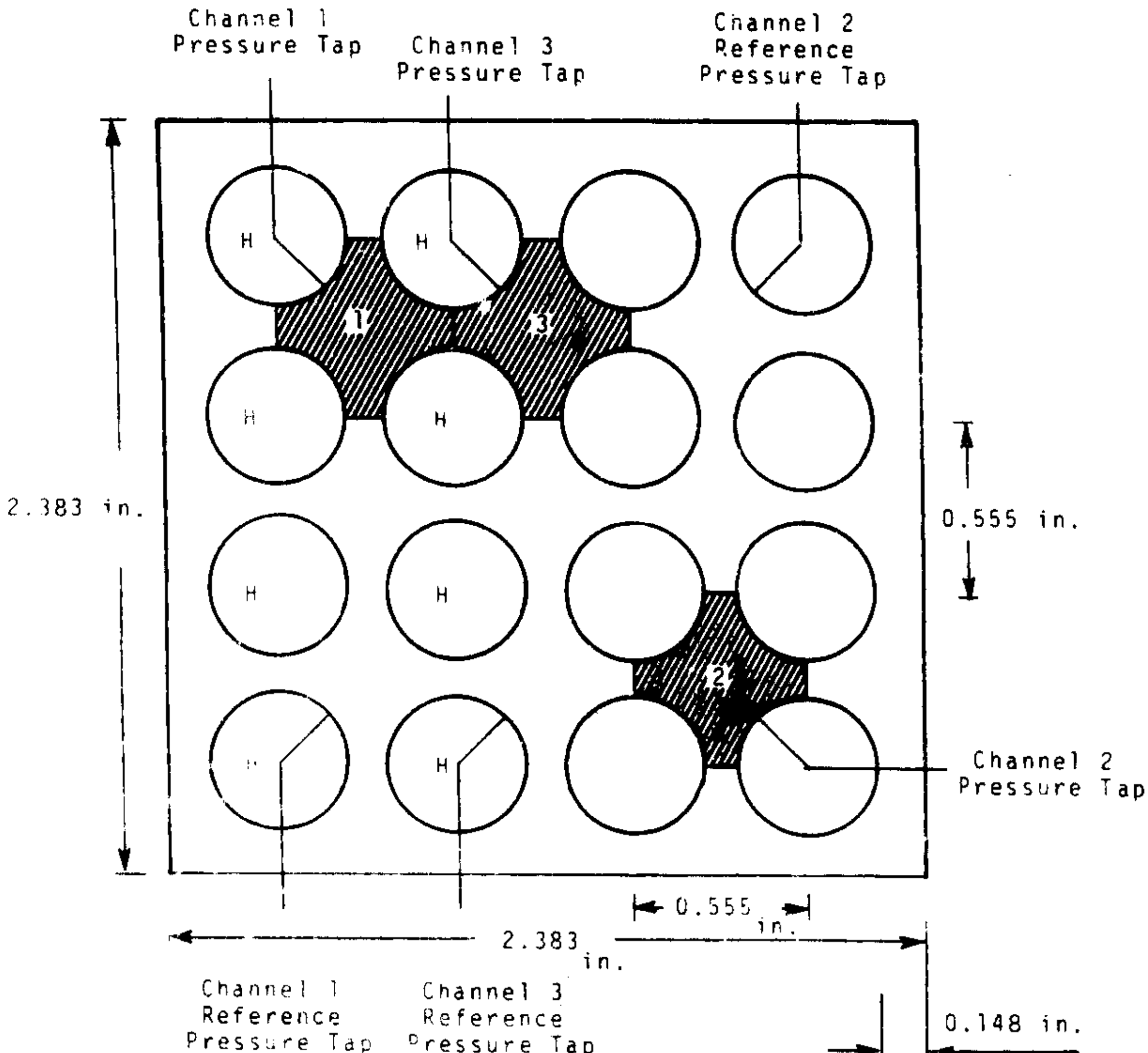
Measurement of Flow and Enthalpy in the Subchannels of a BLW-250 Geometry Rod Bundle - TAC 3.3.9
Columbia University Engineering Research Laboratory
(J. E. Casterline)

Preliminary analysis of flow and enthalpy data in the isolated subchannels of the BLW-250 geometry has been completed. The results have been tabulated, and confidence limits for the experimental measurements established. Where there was scatter in the data, it appeared to result from limitations in the flow measuring instrumentation used to determine subchannel flow rates. The problem was one of instrumentation and will be eliminated in the last series of tests through incorporation of a direct reading frequency counter to measure the turbine flow meter output. It should be noted that errors from this source in no way precluded an effective analysis of the data, since they represented only a small fraction of the variation in subchannel flow. The final phase of analysis, which is currently underway, involves both a complete graphical presentation of the data, and a study in light of the existing computer codes.

The geometry of the U.S. Open Spaced Bundle has been modified as shown in Figure 8. This change was made in order to allow use of a ceramic shroud liner. The asymmetrical geometry previously considered would have required the use of an asbestos-phenolic liner, which has been shown in past tests to be less dimensionally stable and to have an undesirably high friction factor, compared with the ceramic. As indicated in the March HWRP monthly report, the test program will be conducted in two parts. The first will involve simultaneous measurement in Channels 1 and 2, and the second, which will be conducted if time and funding permit, is confined to measurement in Channel 3 alone. The investigation will be carried out at system pressures of 500 and 1200 psia. In previous work, pressures of 500, 700, 1000, and 1200 were investigated. Reduction in the number of pressure levels investigated will allow more time for an intensive study within a wide range of exit quality and heat flux conditions. It is expected that this work will be completed in June.

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

There are two test sections that are being fabricated for this task. The rods in both bundles will be fabricated to simulate short, discrete fuel bundles present in the Pickering reactor (18 in. heated length per bundle with 19.34 in. between simulated end plates).



Rod Outside Diameter	0.422 in.
Rod Pitch	0.555 in.
Rod to Rod Spacing	0.133 in.
Rod to Wall Spacing	0.148 in.
Total Flow Area	0.02389 ft^2
Subchannel Area	0.001168 ft^2
Radial Heat Flux:	
Hot Rods [H]	100%
Cold Rods	86%
Isolated Channels	1, 2, and 3

FIGURE 8. . S. Open Spaced Bundle

The first test section will consist of four 18 in. uniformly heated bundles having a total heated length of 72 in. The second will have five 18 in. bundles and one 15 in. bundle for a total heated length of 105 in. The exit three segments in this latter bundle will be at successively lower heat fluxes to approximate a half cosine. More detailed information on both test sections was presented in the HWRP monthly report for January 1968.

The nickel end connectors and the end plate segments for both test sections have been machined. The tubing has been cut into segments for the first test section, and rod fabrication has started. All other components are currently in fabrication. Experimental operation is scheduled for the latter part of June.

<u>MG Sets for Power Increase</u>

The motor generator sets which were selected to provide increased power capability are presently in Houston, Texas. The windings on all motor generator sets are undergoing a slow baking process prior to reinsulation of the conductors. The specifications, which have been written, require that the dc generators all be separately excited and that a series differential field be wound on the pole pieces. The series differential field must have a voltage drop of 20% from no load to full load. In addition, an adjustable shunt is to be suppli d which will enable adjustment of the external characteristics so that the optimum region can be used for parallel operation over the entire voltage range from 0 to 250 V dc. Additional tests specified for the equipment include insulation tests, no load characteristic, and full load (50% overload) tests. It is anticipated that the equipment will be reconditioned and altered by late August.

It is necessary to provide for the accommodation of the motor generator sets in the building prior to their shipment. The Contracting and Engineering firm of Spencer, White and Prentis is developing an estimate in accordance with Columbia specifications as to base size and location. This firm designed the base of the existing motor-genera or set.

B. STABILITY AND CONTROL

Analysis of Spatial Flux Control - TAC 4.3
Oak Ridge National Laboratory
(John V. Wilson)

During May the principal accomplishments were completion of programs for simulating operation of the Pickering reactor under two new control schemes. Sample calculations for each scheme were sent to AECL personnel for their information. In addition, work continued on developing additional programs to simulate other control schemes, especially the sector control scheme used in the British large gas-cooled reactors.

On May 28 an oral presentation of project progress and plans was made to the Stability and Controls Subcommittee, TAC, at their meeting in Richland, Washington.

C. REACTOR PHYSICS

Temperature Coefficients Measurements - TAC 5.2
Savannah River Laboratory
(B. C. Rusche)

Measurements of the uniform temperature coefficient of reactivity have been performed for a D_2O-cooled lattice of simulated burned-up UO_2 fuel in the SE. The fuel assembly consists of 31 fuel rods with Pu/U weight fraction of 0.00259. The lattice contained nineteen assemblies on a 12.12 in. triangular pitch. Results of the measurements are presented in Table VI.

Material bucklings calculated using the HAMMER code for a D_2O purity of 99.75 mole % are shown in Table VI. Effective radial bucklings for the SE, calculated using the PDQ-05 code, are also given and have been used with the measured values of H^2 to determine the material bucklings. HAMMER calculated D_2O purity coefficients have been used to correct the measured values of H^2 to a purity of 99.75 mole % D_2O. The material buckling values reported have been corrected by adding the experimentally determined immersion heater worth of 19 μb.

TABLE VI. *Buckling as a Function of Temperature*

Material Between Tubes	Temperature, °C	Measured H^2, μb	Moderator Purity, mole % D_2O	$H^{2(b)}$ Corrected to 99.75	PDQ-05 B_R^2, μb	Measured[c] B_m^2, μb	HAMMER B_m^2, μb
Air	23	644	99.61	628	912	284	207
Air	23	668	99.34	621	912	291	207
D_2O	25	674[a]	99.60	656	920	264	201
D_2O	41	674[a]	99.59	657	918	261	-
D_2O	60	672[a]	99.59	655	915	260	203
D_2O	82	680[a]	99.59	654	911	257	205

(a) *Corrected for presence of immersion heater*

(b) *Corrections made using HAMMER calculated moderator purity coefficients*

(c) B_m^2 *determined from PDQ-05 calculated* B_R^2 *and measured values of* H^2 *corrected to 99.75 mole %*

The region between the inner and outer housing can, which is normally filled with air, was filled with D_2O for the temperature measurements to provide an adequate heat transfer medium between the bulk moderator and the fuel. Buckling measurements were made at room temperature for the air-filled case.

Detailed foil activation measurements have been performed at both 23 and 80 °C. Analysis of the data obtained is currently underway.

HAMMER Code - TAC 5.4
Savannah River Laboratory
(H. C. Honeck)

No report was received this month.

D. FUEL AND MATERIAL TECHNOLOGY

Evaluation of Zr--2.5 wt% Niobium Pressure Tubing - TAC 6.9
(F. J. Pankaskie)

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

Crack growth and propagation studies are a part of the over-all program to evaluate Zr—2.5 wt% Nb alloy for pressure tube applications in nuclear reactors. As a part of the crack growth and propagation studies an initial series of tests were completed on tubing in the cold-worked and hydrided condition and the heat-treated and hydrided condition. Results of these tests are summarized in Table VII.

Except for two test specimens (23 and 8) all specimens contained nearly full length longitudinal welds. These test specimens had previously been tested with milled slots.* The cracks, which had originated from the 1/8 in. wide milled slots, were welded in an inert atmosphere so that these same test specimens could be used in fatigue crack growth and propagation tests. In all of the welded test specimens the fatigue crack initiating

---

* P. J. Pankaskie. Crack Propagation Characteristics of Zr-2.5 wt% Nb Alloy Tubing, BNWL-560, Pacific Northwest Laboratory, Richland, Washington. October 1962.

TABLE VII. *Crack Growth and Propagation Behavior of Hydrided Zr-2.5 wt% Nb Tubing*

Specimen Identity		Slot Size, in.	Cycles to Leak	Crack Length at Leak, in.	Cycles to Failure	Crack Length at Failure, in.	Max Hoop Stress, in.	Failure Mode
25	Heat treated and hydrided	0.125	23,765	0.41	25,830	0.46	14,000	Tension
8[a]	Heat treated and hydrided	0.125	-	-	-	0.125[a]	15,000 (a)	Tension
10 A	Heat treated and hydrided	0.125	50,180	0.56	79,120	1.20	17,400	Tension
17 A	Heat treated and hydrided	0.125	18,500	0.36	49,680	0.76	20,000	Tension
21 A	Heat treated and hydrided	0.125	6,470	0.40	6,522	0.42	26,000	Tension
30-8	30% CW Hydrided	0.125	53,850	0.40	86,800[b]	3.05[b]	17,000 (b)	Failure did not occur at fatigue crack
30-11	30% CW Hydrided	0.125	17,900	0.38	47,000	2.65	21,000	Limited ductile tearing
60-11	60% CW hydrided	0.125	23,300	0.41	39,000[c]	1.00[c]	21,000 (c)	Limited ductile tearing

(a) *This specimen failed during the initial pressurizing cycle. The crack propagated the full length of the test specimen.*

(b) *When the fatigue crack had growth to 3 05 in. the specimen was pressurized to failure. In pressurizing, the 3.05 in. fatigue crack extended by ductile tearing to a total length of about 3.7 in. At a hoop stress of 21,000 psi, a failure was initiated in the heat affected zone of the weld and that failure propagated the full length of the test specimen.*

(c) *When the fatigue crack had grown to a length of 1 in. the specimen was pressurized to failure. Failure occurred at a hoop stress of 67,000 psi. Failure occurred by ductile tearing about 1-3/4 in. from each end of the fatigue crack.*

electrical discharge machined (EDM) slot was located at least one-fourth of the circumference away from the longitudinal weld to minimize heat and residual stress effects arising from the weld. For the two test specimens which had not previously been tested, the EDM slot was located at the point of minimum wall thickness at the mid-length of the specimen. All test specimens were hydrided under identical conditions to concentration levels of 200 to 275 ppm corresponding to saturation at 400 °C to achieve substantially uniform hydride throughout the test specimens. For the heat-treated and hydrided specimens, the aging treatment was corrected to compensate for the time at temperature required in the homogenizing treatment. A 10 mil surface layer was removed from the outside diameter surface of the three cold-worked specimens for the purpose of removing any massive hydride layer that may have been present. Except for Specimen 8, there was no removal of surface material from the heat-treated and hydrided test specimens. From the outside diameter surface of Specimen 8, abcut 10 mils were removed in the vicinity of the EDM slot.

Figure 9 shows the curves of hoop stress at failure versus critical crack size for heat treated material in both the unhydrided and hydrided condition size. In Figure 9 it is seen that for the heat-treated and unhydrided condition, the critical crack size in a test specimen with a milled slot is roughly 60% greater than in a test specimen in which a fatigue crack is grown. At the same hoop stress one suspects that this difference is primarily attributable to the sharpness of the fatigue crack and therefore greater stress intensity. There is some data (BNWL-560) that suggests that prior fatigue damage may reduce critical crack sizes.

For the heat-treated and hydrided condition, tubing with a small milled slot appears to be stronger than it would be if a comparable sized fatigue crack were present. With appreciably longer fatigue crack and milled slot sizes, the situation appears to be reversed. The reason for this behavior is not now obvious.

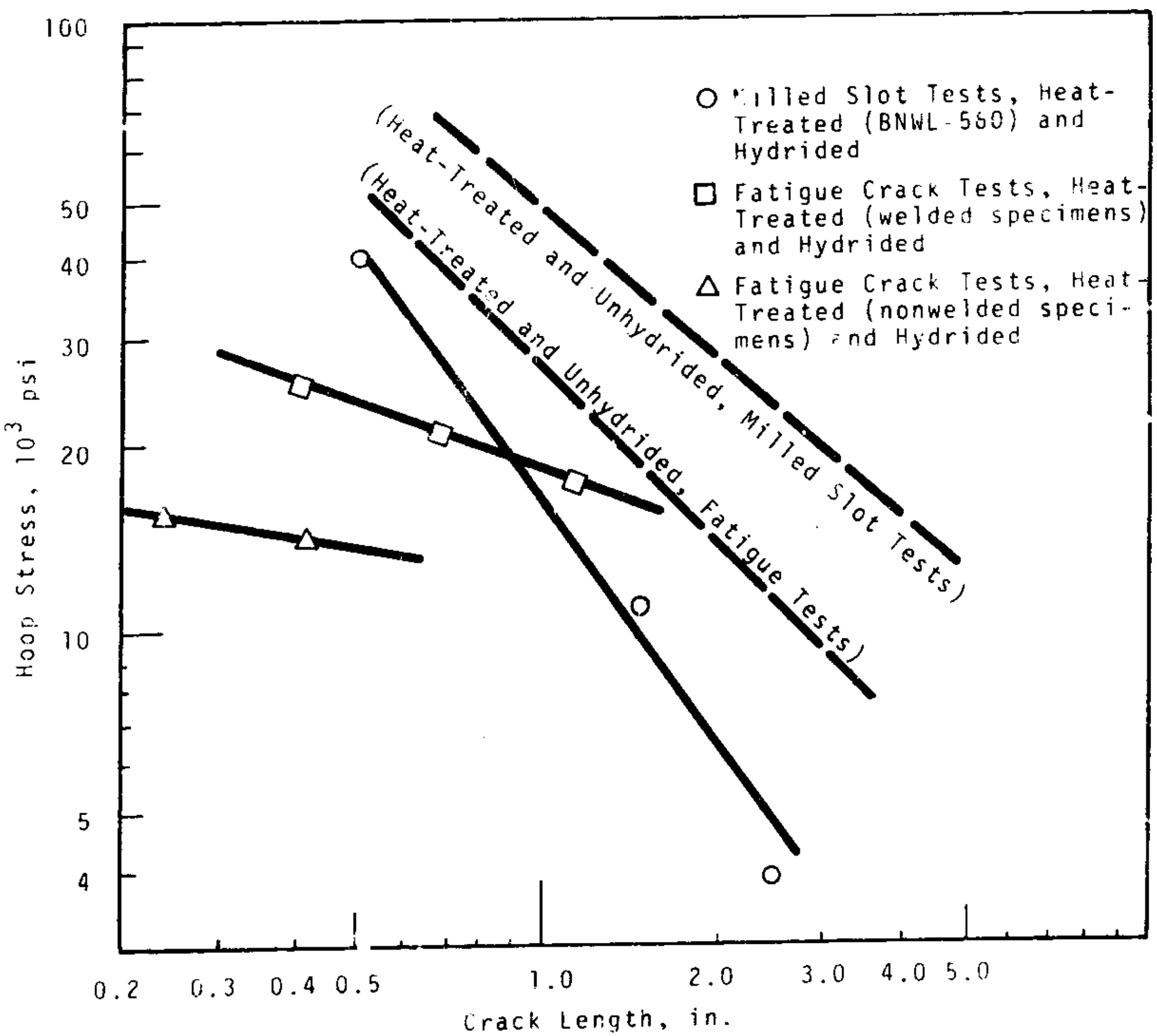


FIGURE 9. Fracture Behavior of Heat-Treated Zr—2.5 wt% Nb Tubing in the Hydrided and Unhydrided Condition as Determined by Milled Slot and Fatigue Crack Tests

All of the welded specimens (10A, 17A, and 21A) had previously been tested at room temperature with milled slot lengths of 4 in., 1/2 in. and 1-1/2 in., respectively. At room temperature, hydride reorientation from inert atmosphere welding would not be expected to alter the total hydride concentration level or singificantly alter the hydride orientation and distribution at locations distant from the weld and heat affected zones. Test Specimens 23 and 8 (see Table VII) had not been subjected to prior tests with milled slots or welding. The fracture strength and critical crack sizes are appreciably less than for those subjected to prior slot tests and subsequent welding. Based on these data, one tends to conclude that either the prior slot test and/or welding is a dominant factor in these observed differences. Metallographic and fractographic examinations are needed to ascertain the structural differences in these two groups of test specimens.

In the case of the three cold-worked and hydrided test specimens (see Table VII) there was no significant crack propagation from the fatigue cracks. Ductile tearing could be induced from the tips of the fatigue cracks only so long as sufficient internal pressure could be maintained. In one specimen, 30-8, crack propagati n did occur. Under internal pressure, a propagating crack was initiated in or immediately adjacent to the heat-affected zone of the longitudinal weld seam even though a 3 in. long fatigue crack was present. This crack was initiated and propagated nearly the full length of the test specimen (16 in.) at a hoop stress of 21,000 psi. This crack was initiated on the side of the weld nearest the 3 in. fatigue crack. An appreciable amount of bulging occurred between the origin of the propagating crack and the 3 in. long fatigue crack. This bulging apparently turned the running crack ends away from the weld zone and toward the fatigue crack ends. As the running crack approached the length of the fatigue crack, it again turned and ran parallel to the tube axis. Hence, while the fracture toughness of the cold-worked and hydrided tubing appears to be much better than for the heat-treated and hydrided tubing, "brittle" crack propagation can, under proper circumstances, occur. It is suspected that the failure originated at some flaw or defect in or adjacent to the heat affected zone of the weld since a similar test specimen was able to withstand a hoop stress of 67,000 psi without failure. Residual welding stresses were probably a contributing factor. Fractographic examinations will be made to ascertain the origin and course of failure.

In order to provide test material for postirradiation crack growth and propagation studies, four tubular test specimens are being irradiated in an unfueled process channel in the PRTR (Test 127). Two specimens are being irradiated in the cold-worked condition and two in the cold-worked and hydrided condition. In this unfueled channel, the fast flux ($E > 1$ MeV) is estimated to be 10^{12} nv. Accumulated exposure is currently estimated at 2×10^{20} nvt.

The cold-worked Zr—2.5 wt% Nb alloy pressure tube operating as a standard fuel channel in the PRTR continued to accumulate exposure (total 2×10^{20} nvt) during the reporting period. No tube inspections were performed, and no diameter measurements to determine creep were made.

Insertion of a second Zr—2.5 wt% Nb pressure tube into the PRTR has been proposed with the primary objective of determining in-reactor creep by periodic diameter measurements and a secondary objective of obtaining operations experience with an unautoclaved tube. The tube, in the heat-treated condition, has been prepared and is ready for use. A detailed stress analysis of the tube, including those stresses encountered during reactor transients at the transitions of the two sections with reduced walls, are within the requirements of Section III of the ASME nuclear code.

The placement of the heat-treated Zr—2.5 wt% Nb pressure tube is being held in abeyance until the questions raised concerning fracture characteristics of this material when in the heat-treated and hydrided condition are satisfactorily resolved.

In-Reactor Measurements of Creep in Zr–2.5 wt% Nb Alloy - TAC 6.10
Pacific Northwest Laboratory
(E. P. Gilbert)

An in-reactor creep test being conducted at a stress level of 52.7 kg/mm^2 at 300 °C is creeping at a rate of approximately 1.5×10^{-6} hr^{-1}. Corresponding unirradiated control creep rates were 4.3×10^{-6}, 4.2×10^{-6}, and 1.3×10^{-5} hr^{-1}. These results agree with earlier tests that indicated that the neutron flux decreases creep rates at high stresses compared to its accelerating effect on creep at low stresses.

In-Reactor Corrosion of Zirconium Alloys - TAC 6.14
Pacific Northwest Laboratory
(A. B. Johnson, Jr.)

Three specimen holders were charged into the G-7 loop of the Engineering Test Reactor at the beginning of Cycle 95 (March 18, 1968). Specimens and goal exposures were summarized in the February 1968 Monthly Progress Report. ETR cycles have typically been approximately 20 days. However, current practice is to attempt to extend the ETR cycle length. The goal exposure (60 days) is expected to be achieved on Quadrant 294 at the end of Cycle 96 (June 24, 1968). It will be discharged and shipped to Battelle-Northwest for postirradiation examination. The quadrant contains corrosion coupons of Zircaloy-2 and Ozhennite in various prefilm conditions.

U_3Si Irradiations - TAC 6.15
Pacific Northwest Laboratory
(R. D. Leggett)

The purpose of this program is to establish the dependence of swelling in U_3Si on temperature, pressure, burnup, burnup rate, and metallurgical state in order to determine the basic irradiation behavior of the fuel. Small specimens will be irradiated in temperature and pressure controlled NaK-filled capsules and the changes in density, hardness, and structure evaluated by detailed postirradiation examination. Sections of fuel rods that have been irradiated by AECL are being examined in order to adjust the present examination techniques to U_3Si fuel material and to allow cross-comparison of results.

Irradiation Program

Two capsules connected in tandem, each containing a pair of U_3Si specimens, are under irradiation at 50 psi and 625 and 725 °C, respectively, for a goal exposure of 0.2 at.% BU. Tandem capsules provide similar irradiation environments and reduce irradiation costs. Construction is continuing on three additional capsules, each containing four U_3Si specimens, designed to operate at 1000 psi to 0.2 at.% BU. Specimen temperatures will range from 350 to 625 °C.

Postirradiation Examination

Quantitative metallography is continuing on sections of three irradiated rods (MJE, 990 MWd/tonne U; MJB, 1620 MWd/tonne U; and MJL, 5900 MWd/tonne U) to determine the size-frequency relationship of the porosity and its contribution to the total volume increase. At least six electron micrographs of the edge and the core of each rod are being analyzed to obtain a representative volume increase. Previous electron microscopy revealed that swelling appears in the form of tears in MJE and as spherical pores in MJL. MJB contained both tears and spherical pores. It is not known at this time whether the pores are filled with fission gas or are merely sintered tears.

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 to 700 °C range to peak exposures of about 12,000 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 Fabrication

Ozhennite sheath tubing for the unbonded fuels was annealed over mandrels to stabilize the metallurgical structure and to straighten the tubing. The annealing conditions of 600 °C and 1 hr are sufficient for full recrystallization. Nondestructive testing of the tubing by eddy current and ultrasonic methods will be completed soon.

Machining of pellets for the unbonded fuels is complete. The pellets are now being inspected. An inspection method was devised for accurately measuring the angle and/or depth of the conical voids that were machined in the pellet ends. The method can be used on irradiated pellets as well.

All coextrusion billets were assembled and extruded. Dimensions appear to be excellent. Very smooth, uniform central voids were produced. The smaller mandrels (approximately 0.120 in. diam for the 5% void) tended to neck off due to two factors, the small cross sectional area and low hardness of the mandrel. The hardness of all mandrels (T-1 tool steel) was about Rc60 instead of the desired Rc63 to 65. The larger mandrels (approximately 0.170 in. diam for the 10% void) performed very well despite original hardnesses of Rc60. Two mandrels were used for all six of the 10% void coextrusions. A drop in hardness of only 3 to 5 points Rc was found. The six 0.120 in. diam mandrels had a similar hardness drop but were used for only one extrusion each.

Metallography is in progress to confirm the dimensions and bond quality of all coextrusions. The coextrusions were cut to fuel lengths and beta heat treated. End cap recesses are now being machined into them.

Development of the electron beam welded end closures is complete. Development of the spacer tab attachment by magnetic force welding is scheduled to begin when the TIG welded end closure is satisfactorily developed.

Work on the gas tungsten arc welded end closure is proceeding. The principle problems are (1) developing a joint design that will permit the free escape of helium from the fuel tube as the heat from welding raises the temperature, and (2) minimizing the temperature rise in the enclosed helium gas. The original approach used six very small projections of precise height on the end of the tube. These produced a uniform annulus between the end of the tube and the end cap. It soon became evident that the gap prevented coalescence between the molten metal on the end cap and tubing. This approach was then abandoned in favor of placing the cap and tube end in contact. A copper chill block was fixed on the tubing in close proximity to the weld zone in order to prevent excessive heating of the helium gas. This arrangement was only partially successful. Problems of either incomplete penetration or blow-out due to gas pressure buildup have prevented obtaining reproducibly sound welds. A vented cap closure is now being investigated.

Capsules were prepared for both sheath thicknesses with internal gap lengths varying from 0.075 to 1.00 in., simulating the void at the end or between pellets. These have been subjected to autoclave tests to determine cladding collapse conditions. No deformation resulted from approximately 8 hr at 300 psig at 850 °F. Tests with increasing pressures at 800 °F showed that the 0.020 in. thick cladding was stable up to 550 psig with collapse in the longest gap length at 600 psig. Additional tests have been completed up to 1000 psig at 800 °F, which still show no dimpling in the 0.035 in. thick sheathing. As expected, several of the gaps less than 1.00 in. are now beginning to collapse in the thinner sheathing. These tests have verified that under the irradiation conditions anticipated for the pellet fuel there should be no problem with clad collapse.

E. D_2O TECHNOLOGY

No report was received this month.

F. SAFETY

Particle Transport to Surfaces by Condensing Steam - TAC 10.1

Pacific Northwest Laboratory
(L. C. Schwendiman, T. W. Horst, A. K. Postma, J. M. Hales)

The review of particle transport in a condensing steam system was brought to its final form by incorporating the most recent literature on thermophoresis and diffusiophoresis and by beginning a detailed study of the nucleation of water drops on particles and their subsequent growth The aim of the latter study is the prediction of the final size of aerosol particles and spray drops due to condensation to them and of the collection of particles by falling spray drops due to the thermal and vapor concentration gradients in their immediate vicinity.

The preliminary theoretical investigation of condensation in steam-air systems progressed to a point where an experimental design is being considered. The proposed experiment will consist of a vertical condensation plate located in a chamber where constant conditions of temperature, pressure, and humidity can be maintained. The plate will measure approximately 3 x 8 ft (large enough to induce a turbulent boundary layer), and will consist of 1 ft segments mounted on a vertical rack.

Temperature and velocity profiles will be measured by hot-wire and thermistor probes. Specifications for a hot-wire anemometer have been prepared and a purchase requisition written.

Individual segments of the plate have been designed so that wear and particle capture by that segment can be determined by a simple, small displacement on the rack. Heat loads to individual segments will be measured by observing temperature rise in the cooling medium.

It is intended that two such plate segments will be built in the near future, and these will be used for preliminary tests of the experimental design.

Interest in the boundary-layer equations describing condensation in laminar-flow systems has led to the application of analog computers for their solution. Circuits are currently being designed for these solutions.

DISTRIBUTION

No. of Copies	
1	AEC Chicago Operations Office 9800 South Cass Avenue Argonne, Illinois 60439 G. H. Lee, Director Patent Division
2	AEC Chicago Patent Group Federal Bldg., Area 700 Richland, Washington 99352 R. K. Sharp
1	AEC Division of Technical Information Extension P. O. Box E Oak Ridge, Tennessee 37830 Head, Division of Technical Information Extension
22	AEC Library, Washington 20545 Division of Reactor Development Technology A. J. Pressesky, Chief Analysis and Evaluation Board E. E. Kinter, Chief Core Design Branch W. A. Williams, Chief Desalting Branch J. W. Crawford, Asst. Dir. Engineering Standards E. E. Kinter, Chief Fuel Fabrication Branch J. M. Simmons, Chief Fuels and Materials Branch A. N. Tardiff, Chief (2) Heavy Water Reactors Branch R. W. Bean, Chief Instruments and Control Branch J. A. Lieberman, Asst. Dir. Nuclear Safety (5) M. A. Rosen, Asst. Dir. Plant Engineering D. E. Pollock, Asst. Dir. Program Analysis A. Giambusso, Asst. Dir. Project Management E. A. Kinter, Asst. Dir. Reactor Engineering I. F. Zartman, Chief Reactor Physics Branch E. E. Sinclair, Asst. Dir. Reactor Technology N. Grossman, Chief Special Technology Branch D. E. Erb, Acting Chief Water Projects Branch

No. of Copies	
1	AEC Library, Washington 20545 Division of Technical Information MS149 W. A. Kee, Technical Utilization Officer
1	Office of General Counsel R. A. Anderson, Asst. General Counsel for Patent
1	AEC RDT Senior Site Representative P.O. Box 550, Area 300, Bldg. 3760, Richland, Washington 99352 P. G. Hoisted/A. S. Waterhouse
1	AEC RDT Senior Site Representative OROO, P.O. Box X Oak Ridge, Tennessee 37830 D. F. Cope
1	AEC RDT Site Representative P.O. Box 2108 Idaho Falls, Idaho 83401 K. A. Trickett
1	AEC RDT Site Representative P.O. Box 500 Windsor, Connecticut 06095 G. U. Merckel
1	AEC RDT Site Office P.O. Box 591 Canoga Park, California 91305 Robert L. Morgan, Senior Site Rep., AI
2	AEC Richland Operation Office Federal Bldg. Richland, Washington 99352 C. L. Robinson, Director Research and Development Division Technical Library
3	USAEC Scientific Representative c/o Atomic Energy of Canada Ltd. Chalk River, Ontario, Canada R. W. Ramsey

No. of Copies	
1	USAEC Technical Representative c/o Atomic Energy of Canada Ltd. Sheridan Park Ontario, Canada
	H. J. Reynolds
	USAEC/AECL COOPERATIVE PROGRAM TAC AND SUBCOMMITTEES
5	Technical Advisory Committee (TAC) - United States
	A. N. Tardiff, U.S. Chairman of TAC AEC Headquarters Washington, D. C. 20545
	H. J. Reynolds, U.S. Secretary of TAC c/o Atomic Energy of Canada Ltd. Sheridan Park, Ontario, Canada
	D. F. Babcock, U.S. Member of TAC E. I. duPont de Nemours & Co. Wilmington, Delaware
	H. Harty, U.S. Member of TAC Heavy Water Reactor Program Office Battelle-Northwest Richland, Washington 99352
	F. A. Mason, U.S. Member of TAC Massachusetts Institute of Technology Cambridge, Massachusetts, 02139
5	Technical Advisory Committee (TAC) - Canadian
	D. D. Stewart, Canadian Chairman of TAC Atomic Energy of Canada Ltd. Chalk River Nuclear Laboratories Chalk River, Ontario, Canada
	W. R. Livingston, Canadian Secretary of TAC Atomic Energy of Canada Ltd. Chalk River Nuclear Laboratories Chalk River, Ontario, Canada
	L. Pease, Canadian Member of TAC Atomic Energy of Canada Ltd. Toronto, Ontario, Canada

No. of Copies	
	R. F. S. Robertson, Canadian Member of TAC Atomic Energy of Canada Ltd. Whiteshell Nuclear Research Establishment Pinawa, Manitoba, Canada
	W. R. Thomas, Canadian Member of TAC Atomic Energy of Canada Ltd. Chalk River Nuclear Laboratories Chalk River, Ontario, Canada
7	TAC Subcommittee - REACTOR STABILITY AND CONTROL
	R. Feit Division of Reactor Development & Technology U.S. Atomic Energy Commission Washington, D. C. 20545
	A. Pearson Atomic Energy of Canada Ltd., Chalk River Nuclear Laboratories Chalk River, Ontario, Canada
	F. Bevilacqua Combustion Engineering, Inc. P.O. Box 500 Windsor, Connecticut 06095
	G. Dessauer E. I. duPont de Nemours & Co. Savannah River Laboratory P.O. Box 117 Aiken, South Carolina 29801
	L. Pease Atomic Energy of Canada Ltd. Toronto, Ontario, Canada
	D. B. Primeau Physics & Analysis Branch Atomic Energy of Canada Ltd. Sheridan Park, Ontario, Canada
	J. R. Worden Engineering Development Dept. Battelle-Northwest Richland, Washington 99352

No. of Copies		No. of Copies	
4	TAC Subcommittee - HEAT TRANSFER AND FLUID DYNAMICS R. M. Scroggins Division of Reactor Development and Technology U.S. Atomic Energy Commission Washington D. C. 20545 G. A. Wikhammer Advance Engineering Branch Advanced Project & Reactor Physics Division Atomic Energy of Canada Ltd. Chalk River, Ontario, Canada J. M. Batch Engineering Development Dept. Battelle-Northwest Richland, Washington 99352 J. S. Neill E. I. duPont de Nemours & Co. 1007 Market Street Wilmington, Delaware 19801		Manager Water Reactor Safety Program Office Phillips Petroleum Co. Idaho Falls, Idaho 83401 J. J. Billington Engineering & Physics Dilworth, Secord, Meagher & Associates 4195 Dundas St., West Toronto, Ontario, Canada J. D. Sainsbury Atomic Energy of Canada Ltd. Sheridan Park, Ontario, Canada
9	TAC Subcommittee - SAFETY A. J. Pressessky Analysis & Evaluation Branch Division of Reactor Development and Technology U.S. Atomic Energy Commission Washington, D. C. 20545 G. Hake Atomic Energy of Canada Ltd. Chalk River Nuclear Laboratory Chalk River, Ontario, Canada H. Morewitz Atomics International P.O. Box 309 Canoga Park, California 91305 R. K. Hilliard Battelle-Northwest Richland, Washington 99352 F. C. Boyd Atomic Energy Control Board Chalk River, Ontario, Canada S. Davies Canadian General Electric Co., Ltd. 107 Park Street, N. Petershorough, Ontario, Canada	6	TAC Subcommittee - FUELS AND MATERIALS A. Van Echo Division of Reactor Development & Technology U.S. Atomic Energy Commission Washington, D. C. 20545 J. A. L. Robertson Atomic Energy of Canada Ltd Chalk River Nuclear Laboratories Chalk River, Ontario, Canada R. G. Wheeler Engineering Materials & Mechanics Section Battelle-Northwest Richland, Washington 99352 W. R. Thomas Atomic Energy of Canada Ltd. Chalk River Nuclear Laboratories Chalk River, Ontario, Canada M. F. Lyons Fuels Development and Design Materials Nuclear Technology Department General Electric Company Pleasanton, California 94566 W. B. DeLong Atomic Energy Division E. I. duPont de Nemours & Co. Wilmington, Delaware 19898

No. of Copies	
8	TAC Subcommittee - PHYSICS J. W. Lewellen, Reactor Physics Branch Division of Reactor Development & Technology U. S. Atomic Energy Commission Washington, D.C. 20545 C. H. Millar Atomic Energy of Canada Ltd. Chalk River Nuclear Laboratory Chalk River, Ontario, Canada J. L. Crandall Experimental Physics Division Savannah River Laboratory E. I. duPont de Nemours & Co. Aiken, South Carolina 29801 G. O'Neill Savannah River Laboratory E. I. duPont de Nemours & Co. P.O. Box 117 Aiken, South Carolina 29801 S. Visner Combustion Engineering P.O. Box 500 Windsor, Connecticut 06095 L. Pease Atomic Energy of Canada Ltd. Chalk River Nuclear Laboratories Chalk River, Ontario, Canada L. C. Schmid Battelle-Northwest Richland, Washington 99352 A. Pasanen Atomic Energy of Canada Ltd. Sheridan Park, Ontario, Canada
1	AAEC Research Establishment Lucas Heights, Australia M. M. Yarosh (To be opened by addressee only)
1	Aerojet General P.O. Box 1845 Idaho Falls, Idaho 83401 W. E. Nyer
1	Babcock and Wilcox Company P.O. Box 1260 Lynchburg, Virginia 24505 M. W. Croft

No. of Copies	
1	Columbia University 632 West 125th Street New York, N.Y. 10027 J. F. Casterline Engineering Research Laboratory Department of Chemical Engineering
-	Combustion Engineering P.O. Box 500 Windsor, Connecticut W. P. Chernock R. C. Kern
1	Douglas United Nuclear Richland, Washington 99352 Raymond Cooperstein
1	E. I. duPont de Nemour & Co. Wilmington, Delaware 19898 Reactor Engineering Section
1	Massachusetts Institute of Technology 77 Massachusetts Avenue Room 24-109 Cambridge, Massachusetts 02139 Department of Nuclear Engineering
2	Oak Ridge National Laboratory P.O. Box X Oak Ridge, Tennessee 37830 J. A. Lane, Studies and Evaluation Reactor Program J. A. Parsons Nuclear Desalting
1	Phillips Petroleum Co. P.O. Box 2067 Idaho Falls, Idaho 83401 Manager, Water Reactors Safety Program Office
2	Savannah River Laboratory Aiken, South Carolina 29801 J. L. Crandall Experimental Physical Research Reactor Engineering S. W. O'Rear, Supervisor Technical Information Services

No. of Copies

1 Whiteshell Nuclear Research Establishment
Pinawa, Manitoba, Canada

R. S. Hart

57 Battelle-Northwest
Richland, Washington 99352

F. W. Albaugh
E. R. Astley
J. A. Ayres
J. M. Batch
A. L. Bement
J. B. Burnham, Jr.
S. H. Bush
J. J. Cadwell
F. G. Dawson
D. R. deHalas
R. E. Dickerson
R. L. Dillon
E. A. Eschbach
E. A. Evans
J. C. Fox
R. S. Hammond
H. Harty
R. K. Hilliard
R. L. Knecht
B. B. Lane

Battelle-Northwest (contd)

G. A. Last
R. D. Leggett
W. R. Lewis
R. P. Marshall
J. E. Minor
P. G. Parr, Jr.
R. S. Paul
L. T. Pedersen
W. W. Porath
W. D. Richmond
D. S. Rowe
L. C. Schmid
D. H. Stewart
L. C. Schwendiman
H. A. Taylor
R. G. Wheeler
O. J. Wick
N. G. Wittenbrock
W. C. Wolkenhauer
J. R. Worden
D. C. Worlton
J. R. Young
F. R. Zaloudek
J. G. Zwiener
Technical Information Files (3)
Legal (2)
HWR Information Center (6)
Technical Publications (2)

END