The Tripartite
ORGANIC COOLED HEAVY WATER
REACTOR MEETING

SANTA MONICA, CALIFORNIA

JUNE 11-13, 1962

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The Tripartite

ORGANIC COOLED HEAVY WATER

Reactor Meeting

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Atomic Energy of Canada Limited

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United States Atomic Energy Commission

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CANADA'S ORGANIC COOLED REACTOR PROGRAM

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SYNOPSIS

Canada's organic cooled reactor program is reviewed
and the most important development projects are discussed.
One of the most important of these projects is on fuel element
fouling. Considerable progress has been made and there is a
good prospect of this being reduced to an acceptable limit.
Materials still present many problems and limit the use of
certain design concepts. However, the SAP quality and avail-
ability is improving and there is real promise of being able to
use zirconium alloys. These latter alloys could simplify the design
and reduce the cost considerably.

The Canadian program began to take form in 1958 when the Canadian
General Electric Co. carried out a design study for a 150 MWe power plant
using a heavy water moderated, organic liquid cooled reactor. The cost
estimates looked attractive so a development program and further design
studies were initiated. CGE has been the primary AECL contractor and the
work has been divided roughly into design studies and out-of-reactor develop-
ment by CGE (Peterborough) and in-reactor work by AECL (Chalk River).
Other contractors such as Orenda Engines Ltd. (Toronto) and General Impact
Extrusions (Toronto) have done certain phases of the development work.
Economic studies have been carried out in parallel with the design and
development work at both Peterborough and Chalk River, and they have continued
to show a gain over CANDU costs which in Canada provide the base target.

As the development progressed consideration was given to a pilot plant
as the next step and in April 1961 a design study was completed for a 30 MWt
reactor called OCDRE. While the study was going on plans were being made
for Canada's new nuclear research center, the Whiteshell Nuclear Research
Establishment, where the OCDRE would be built. Since it would be the only
reactor at Whiteshell for an initial period, it was desirable to incorporate a
number of experimental features in it. This modified some of the original
concepts and as a result some of the basic reactor features were changed, and
the OTR (Organic Test Reactor) was born in late 1961. This reactor has
recently been renamed WR-I. This is being designed now and will be built
and commissioned by AECL. The target date for completion is the end of 1964.
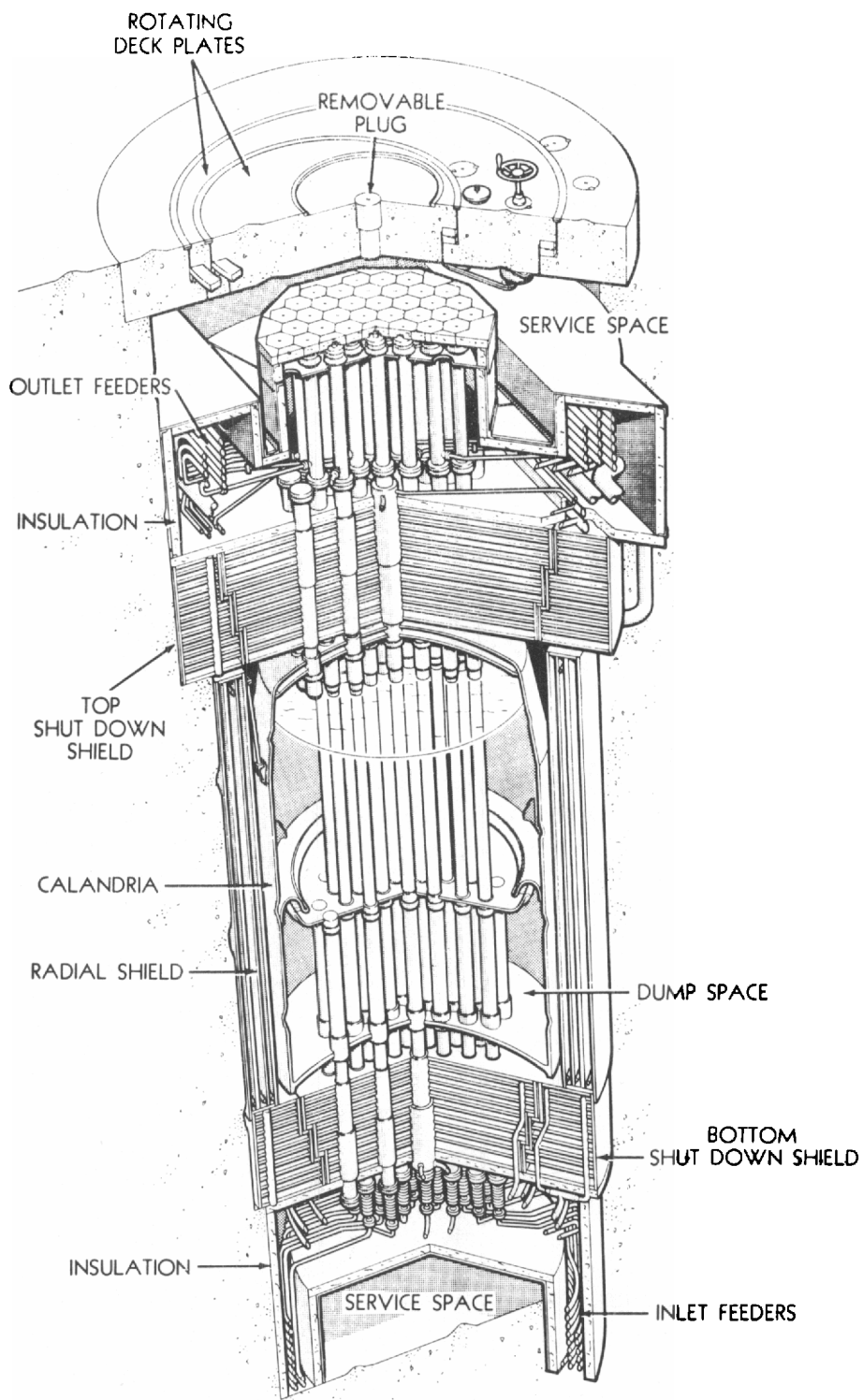
Figure 1.   CUT-AWAY OF WR-I

The basic aim of the WR-I is to test out fuel, materials, and coolants under reactor conditions. A cutaway drawing of the reactor is shown in Figure 1. Those who are familiar with NRX will recognize the similarity. The core consists of a calandria for the cool heavy water, aluminum calandria tubes and hot coolant tubes initially of stainless steel. The fuel is to be made up of 8 ft. long 19-element bundles, the elements to be UO_2 clad in SAP. Each coolant channel has separate inlet and outlet feeder pipes which are joined to headers outside the primary shielding. Initially, provision will be made for two separate coolant circuits each feeding half the coolant channels. Single channels can also be connected separately as normal loops. This core arrangement provides the flexibility needed for testing first on a small scale and then in stages, if required, up to a full reactor scale.

TARGET SPECIFICATIONS FOR OCDR

The basic target, of course, is for a plant which will produce power at a lower cost than any competing system. In order to provide initial goals for the design and development programs a number of initial target conditions shown in Table 1 were selected.

TABLE 1

OCDR Target Conditions

Coolant Outlet Temperature	-	425 °C
Maximum Sheath Temperature	-	510 °C
Surface Fouling Rate	-	<0.001 in./yr.
Coolant Make-up Cost	-	<0.2 mills/kw. hr.
Fueling Cost	-	1 mill/kw.hr. (a burnup of about 8000 MWD/T)

Some of these factors are somewhat interdependent.

THE CURRENT PROGRAM

The effort on the organic program can be divided into three parts; WR-I design and construction, OCDR optimization studies and OCDR development. The effort being expended on the different phases of each of these areas is shown in Table 5.

1. **WR-I Design and Construction**

   Since it is important to have this reactor operating at an early date, it was decided that existing technology would be used wherever possible and that new features would only be incorporated if they could be proved out in time to meet the schedule. CGE are doing all the design and have it well underway. Building construction is due to start mid 1962 and installation of process equipment late in 1963. As mentioned above, the target is to have the reactor completed by the end of 1964. The design of the loops has not started yet so the first loop will probably not be ready until early 1965.

2. **OCDR Optimization Studies**

   These studies are being carried out by both CGE and AECL and at the present time the methods are being compared to resolve any minor differences. These studies will be continued to give guidance to the development programs.

3. **OCDR Development**

   (a) Coolant Decomposition -

   In Canada we have selected the OM terphenyl mixture rather than OMP due to its lower melting point, and all our work has been with the former material. A careful evaluation of the effect of fast neutrons and γ rays on the decomposition rates of terphenyls has been a major problem, most of which is being done in heated X-rod and J-rod positions in NRX. Present indications are that the fast neutrons are 2 - 3 times more damaging than electrons for the same absorption. However, calorimetry has proved to be difficult and it is only recently that this has been worked out to provide satisfactory accuracy in the G values. This work is continuing. Another study is being done to find out how the G-coolant varies with the amount of irradiation and temperature. Electron irradiation from a Van de Graaff is being used. Figure 2. shows the data that have been obtained to date. At higher irradiation doses the G values are found to vary as the square of the amount of undecomposed coolant left. A third phase of the coolant studies is concerned with the basic mechanisms whereby organic molecules decompose under irradiation. Cyclohexane alone and with added benzene is being irradiated and both gaseous and high-boiling decomposition products are being determined. The results to date indicate that at low benzene concentrations a quenching mechanism is present, i.e. the total decomposition is decreased by the benzene. Further evidence for this has come from work with deuterated hydrocarbons. Now work is being done at present on pyrolysis although this is planned as soon as manpower is available.
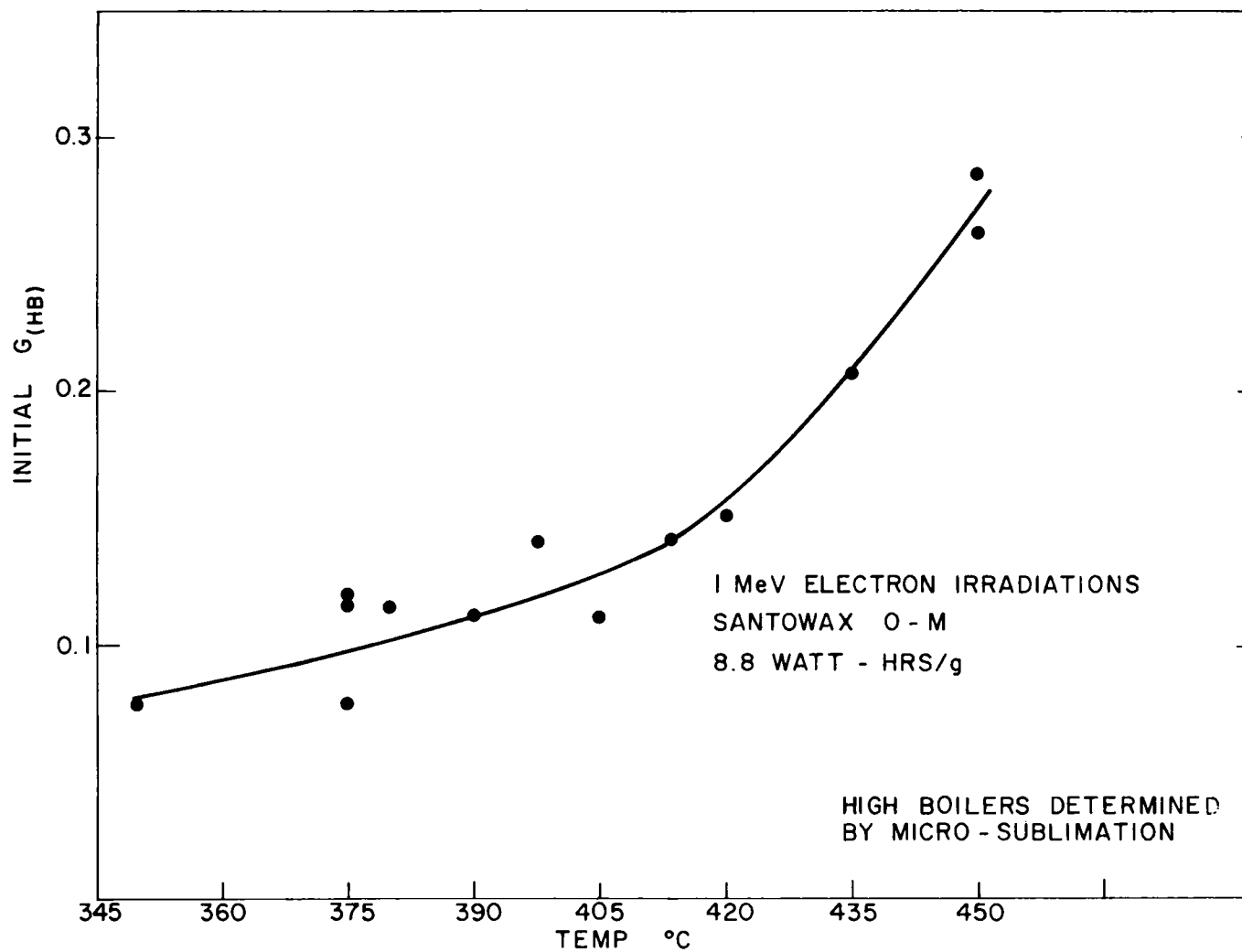
Figure 2.

(b) Fouling -

This work is being done at Chalk River in both out-reactor and in-reactor facilities. In the out-reactor loop the coolant is circulated through purification beds and then through resistance heated stainless steel test sections. The surface temperatures of these sections can be measured to give a continuous measure of the fouling. A number of runs have been completed and the films have been analyzed for iron compounds, ash, carbon, etc. The results to date show:

(1)   There is a good correlation of fouling rate with temperature as shown in Figure 3.

(2)   Purification of the coolant through a bed of Attapulgus clay reduces fouling.

(3)   Increasing the water content of the coolant over the range 40 - 300 ppm increases fouling.

(4)   High velocity films are thicker than low velocity films.

(5)   The heat transfer coefficient can increase and may even double due to a fouling deposit.

(6)   The films contain over 75% iron and iron oxides and some organic material.

All the observations point to mass transfer of iron as being responsible for most or all of the fouling, but the mechanism by which this iron is transported from the cool to the hot surfaces is still unknown. Experiments are also underway to see if the deposits can be cut down by coating the iron in the cooler regions with aluminum. No results have been obtained so far.

All the results to date have been obtained with synthetic coolant mixtures and there is a real need to check these with coolant from a reactor and containing all the decomposition products.

The fouling behaviour of the in-pile fuel irradiations are being carefully followed, but as yet there is no evidence of any large effect of irradiation on fouling.

Coolant cleanup procedures are being evaluated in parallel with the fouling studies. Design data are being collected for the Attapulgus clay columns and for molecular sieve drying columns. It is planned to try and determine what impurities are taken out by the clay but this work has not started.
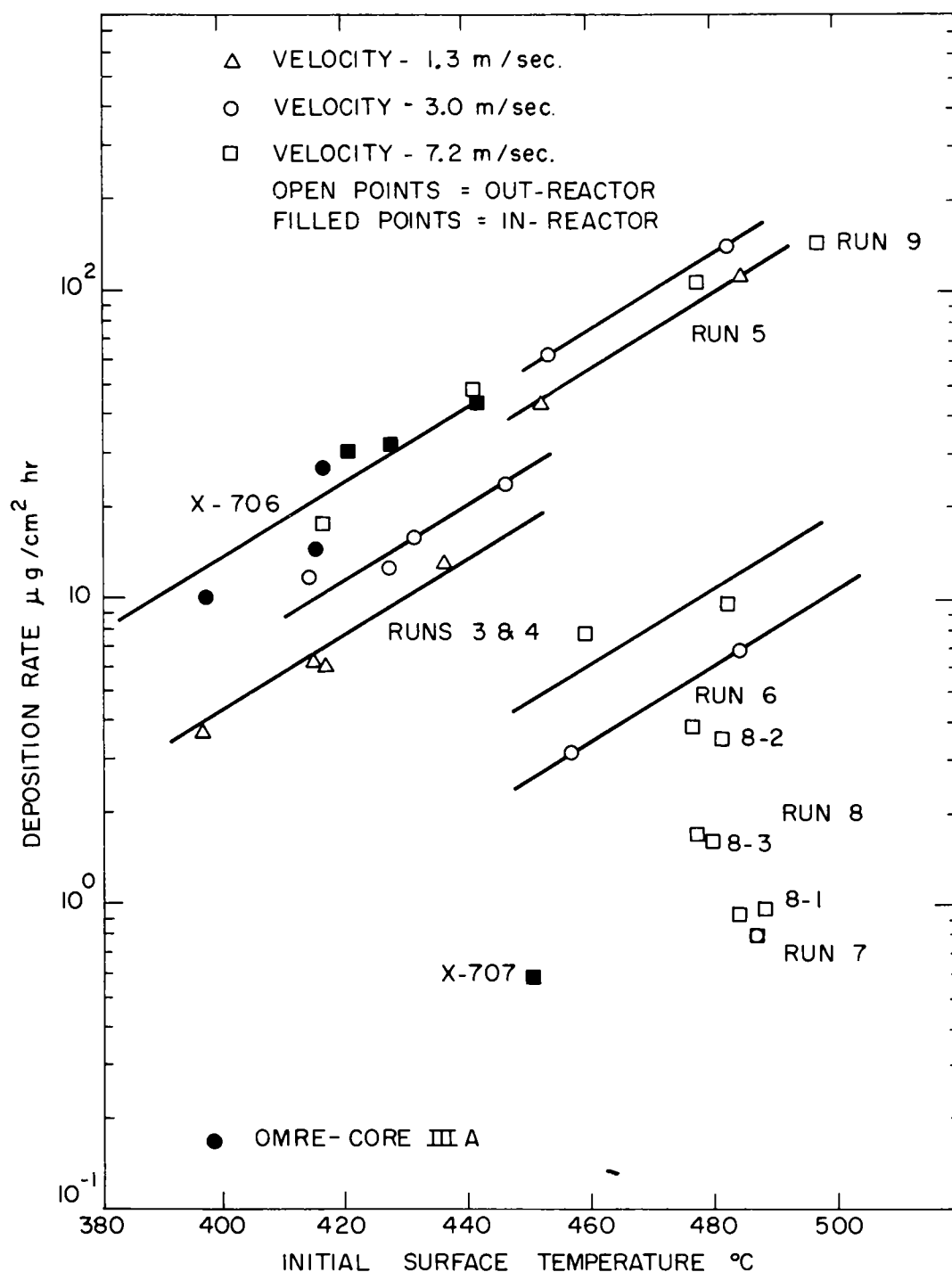
Figure 3

(c)   Heat Transfer -

Most of the organic heat transfer work is done by CGE at Peterborough.
The facilities used for this work are shown in Table 2.

TABLE 2

Heat Transfer Facilities at CGE

	Max. Temp. °C	Pump Head ft. of liquid	Max. Press. psig	Flow USgpm	Test Sections
Pool Boiler	450	-	600	stirred	1
SOTR	450	300	300	25	1
HTOTR	425	450	500	120	2 in parallel

Most of the effort to date has been on burnout measurements. With
the existing correlations the data are badly scattered so this work is
being done to obtain a more accurate correlation which will allow the
maximum heat fluxes to be used in reactor designs. Screening tests
are being done first in the pool boiler, and then dynamic burnout
measurements are being done in SOTR and HTOTR. The range of
conditions used and the experiments planned are shown in Table 3.

TABLE 3

Burnout Program at CGE

Variable	Measured Range	Planned Range
Velocity	5 - 35 ft./sec.	5 - 35 ft./sec.
Sub-cooling	40 - 150°C	5 - 150°C
Composition	None	(Water - 50 - 1000 ppm (L.B.'s - up to 2% (Gas - up to 500 ml/kg
Geometry	None	(Spacing - 0.040 in. (- 0.060 in. (- 0.030 in. (l/d ratio

In general the data follow the Core and Sato correlation.

An important requirement for these burnout studies is an accurate knowledge of vapour pressure and an understanding of how the vapour pressures of the different components affect burnout. This has not re·ceived adequate attention to date. A fundamental investigation of this aspect is being carried out at Chalk River in which the burnout properties of two-component mixtures such as benzene and toluene are measured. The results are showing some peculiarities which have not been explained.

All the current work has been with simple annular geometries where the inner tube is heated. The results may be different in a bundle of elements so equipment is being built at Peterborough to study burnout in a simplified rod bundle geometry. Provision is being made to change the element-to-element spacing, since there is a real economic incentive to use the closest spacing possible. The first results should be available by early June.

(d)   Materials and Corrosion -

SAP is the first choice for core components. Since this is not a commercial material, procurement and product uniformity are the most important problems. Such problems as iron inclusions, high hydrogen content, and non homogeneity of oxide distribution have been encountered and are being overcome by the raw material producers in the USA and in Europe but progress has been fairly slow.

In Canada we have done most of our work with the 7% oxide material but we will probably shift to the 10% grade, as soon as it is available. Mechanical properties such as tensile strength, creep, and stress rupture are being measured by CGE at Peterborough and by Orenda Engines at Toronto. An unexpected effect of cold work which may lead to greater ductility at the operating temperature has been discovered and is being investigated. This may be a very important finding, since low ductility is one of the most serious drawbacks of SAP. Current data indicate that a design stress of about 4000 psi can be used at 400°C for 7% oxide SAP. Development work on the fabrication of fuel sheathing is being carried out in Toronto by the General Impact Extrusion Co. Good quality plain tubing is being produced but spiral ribbed tubing does not have an accurate enough pitch for the current fuel designs. At present the ribbed tubing is being made by machining spiral ribs on plain thick walled tubing. The cost of this operation is being followed carefully but we do not have an accurate estimate yet.

The welding of endcaps to sheaths is being studied at Peterborough. A hot pressure bonding method was worked out and while the bond was good it seemed too slow for production. Following the excellent Hanford results with magnetic force welding, equipment for this operation is being purchased and will be ready to use early in the fall.

The corrosion of SAP in organics has not been studied to date in Canada. However, CGE are setting up a flowing system to study this property at temperatures up to 475°C in the presence of varying amounts of water. While corrosion is not thought to be a problem, this work should provide assurance for its use as fuel cladding under long term exposure to organics.

Zirconium alloys were considered in the early days of the organic program in the USA but were discarded due to unfavourable corrosion results. However, re-examination of the data showed a wide scatter in the results with some indication of promise under certain conditions. Over the past two year period several in-reactor and out-reactor tests have been done at Chalk River with promising results. In the last and most severe test two fuel elements of UO_2 clad in Zircaloy-2 were irradiated for 3 weeks in an organic coolant with sheath temperatures of 460°C. No visible change occurred and a detailed examination showed that the sheaths had picked up essentially no hydrogen. This makes the use of zirconium alloy sheaths and pressure tubes look very attractive, although there still is a great deal of work to be done.

Hydrogen pickup is still considered to be the major problem, particularly with hot coolant tubes which have to remain in the reactor for say 15 years. An aluminum skin on the Zircaloy would cut down hydrogen absorption but a 400°C and above the aluminum would soon diffuse into the Zircaloy. A study of barriers is being made to prevent this. Nickel and chromium and niobium layers are being put on Zircaloy at Orenda Engines and then coated with aluminum. These will be corrosion tested in organic autoclaves at Canadian Westinghouse Co. Another barrier which will be tried is a zirconium oxide layer formed by anodizing.

Several zirconium alloys are being considered but the one with the best high temperature properties is Zr-1.5Al - 0.5Mo; at 400°C it has a working strength of about 13000 psi. Pressure tube samples are being made up and tests in organic with and without barrier layers and coatings will begin shortly. Autoclave tests at Canadian Westinghouse and in-reactor tests at Chalk River will be done.

(e)   Fuels -

Fuel work is divided between CGE and AECL - with CGE doing fabrication development and the preparation of irradiation specimens, while AECL does the irradiations, post irradiation examinations, interpretations and any more fundamental studies. The reference fuel design for the first WR-I charge is a 19-element bundle of SAP clad UO_2 8 ft. long with stainless steel end plates. CGE have made satisfactory end closures

and, as noted above, they are setting up production type equipment. The method for assembling the elements into bundles has not been worked out yet. When a full scale bundle has been assembled mechanical tests, pressure drop and mixing measurements will be done. This work will be carried out later this year.

The fuel irradiation testing at Chalk River is currently being carried out in the X-7 loop in NRX, but a larger loop capable of irradiating a full scale fuel bundle is being designed and built. This is the E-12 loop for NRU and it is due to take the first fuel charge in March 1963. The details of the two loops are shown in Table 4.

TABLE 4

In-Reactor Organic Loops at Chalk River

	X-7 NRX	E-12 NRU
Coolant		
Max. Temperature	427°C	427°C
Max. Pressure	300 lb./in.^{2}	600 lb./in.^{2}
Max. Flow	30 gal.(US)/min.	600 gal.(US)/min.
Test Section		
Inside Diameter	1.5 in.	3.25 in. initially 3.75 in. future
Length	10 ft.	10 ft.
Heat Removal Capacity	200 kw.	2000 kw.

Nine X-7 loop irradiations with UO_2 fuel have been completed, several of which have been cooperative experiments with the USAEC. The operating conditions were close to those expected for WR-I, outlet coolant temperatures 350 - 395°C, maximum sheath temperatures up to 480°C, coolant velocity 15 - 30 ft./sec., irradiation time up to 12 weeks, $\int kd\theta$ of 5.8 - 42.5 W/cm. The results have been very encouraging and only two factors showed up which were significantly different from CANDU fuel. One factor was the lack of ductility of the SAP sheath.

One failure can be attributed to it, but this can probably be overcome by the use of better quality SAP and by some adjustment in the pellet-to -cladding diametral clearance. The second difference was with defected elements. Here the activity release from a defect is much less than from a corresponding defect in a water system, since the organic seals the UO_2 surfaces with a relatively impervious layer of decomposed organic material. On the other hand, a defected element can only stand a limited number of thermal cycles before liquid logging and coke buildup inside the element ruptures the sheath. We have not established the limiting number of cycles or the important variables affecting this, but about 6 cycles may be the limit.

Future irradiation plans for WR-I fuel call for a three month irradiation starting late in May primarily to check fouling, and a one month test to check a fin-on-fin type of bundle design. This will be followed by a seven month irradiation of three 8 ft. long prototype elements of WR-I rating starting in late September. This last irradiation will provide experience with long elements and a long irradiation. The first irradiation in E-12 beginning next March will be with a prototype WR-I fuel bundle.

The second phase of the fuel program is aimed at an advanced fuel for later WR-I charges or for an OCDR. Uranium monocarbide is one such fuel since economic studies show it to have a real advantage over UO_2. A development contract with Eldorado Mining and Refining Ltd. at Port Hope to work out the techniques for producing UC rods has just been started. This work will run for 6 - 12 months and will provide a firmer cost estimate for UC fuel.

Meanwhile at Chalk River we have carried out one joint UC irradiation with Euratom in X-7. The irradiation of SAP-clad and stainless steel -clad UC lasted for 10 days, although defects showed up as soon as the test started. The test was terminated by an increase in pressure drop which was later found to be due to a complete break across one element. This test showed that (1) the break was primarily due to a sheathing defect and the low ductility of SAP, (2) endcap welding must be of a high quality and (3) UC clad in SAP is a promising fuel.

Another experiment using stainless steel clad UC having internal thermocouples was started late in April. The objective here is to get information on the thermal conductivity of UC under irradiation and to measure the interfacial heat transfer between sheath and fuel. The irradiation will be for 3 weeks but no results are available yet.

Other more advanced designs are being considered but are still in the paper assessment stage. These include UC sheathed in impervious graphite, and a seven rod cluster of elements each inside its own pressure tube.

(f)   Coolant Tube Development -

Three designs are being actively worked on and a fourth has been given some consideration. The mechanical development work is being done at CGE while most of the material development is being done at Chalk River. The four designs are shown in Figure 4.

   (i)   Internal Liquid Insulation

        The design is simple mechanically. On the other hand, the semi stagnant annular organic layer absorbs an appreciable number of neutrons and the heat loss is fairly high. For these reasons it is not considered to have a long term future but may be a useful backup design. The question of fouling and plugging of the annulus under operation has not been answered so a test section for the X-7 loop is being designed and may be installed in September for a 7-month period. This will not be long enough to prove out the design but it will be sufficient to proceed with a longer E-12 or WR-I test if desired. Some further out -reactor development would be required before such a test could be designed but this will not be done at present.

   (ii)   Internal Solid Insulation

        Here the insulation is a thin (0.10 in.) layer of insulating material such as Min-K between the hot SAP liner and the cool Zircaloy pressure tube. The insulating value and neutron economy are both good but the mechanical design is difficult due to the severe thermal stresses set up on thermal cycling. Several 3 ft. long models and one 10 ft. prototype have been made and thermally cycled many times from 20° to 375°C. Failures of the liner have occurred so at present this design does not look promising. Calculations show that if the SAP liner is replaced with a Zircaloy liner the stresses are greatly reduced and failure probably will not occur. The problem, however, is then the resistance of Zircaloy to hot organic coolant, which has not been proven to be acceptable yet. Another possibility that is being explored is to use a weak aluminum alloy liner which is plastic at the working temperature.

INTERNAL
LIQUID INSULATION

INTERNAL
SOLID INSULATION

INTERNAL
GAS INSULATION

EXTERNAL
INSULATION

PRIMARY
COOLANT

PRIMARY
COOLANT

PRIMARY
COOLANT

PRIMARY
COOLANT

MECHANICAL
JOINT NOT
LEAK TIGHT

LEAK TIGHT
JOINT

INERT
GAS

INTERNAL
LIQUID
INSULATION

INTERNAL
SOLID
INSULATION

GAS
INSULATING
LAYER

CALANDRIA
WALL

AIR GAP
INSULATION

HOT
UNSTRESSED
LINER

HOT
UNSTRESSED
LINER

HOT
LINER

HOT PRESSURE
TUBE

COOL
STRESS
TUBE

COOL
STRESS
TUBE

COOL
STRESS
TUBE

LEAK
TIGHT
JOINT

GAS
BUBBLES

(a)

(b)

(c)

(d)

Figure 4.

(iii)   Internal Gas Insulation

A gas layer provides the insulation. Some preliminary work
has been done in connection with a test section for the E-12
loop but at present it has been shelved. The design as shown
is relatively simple but the gas which bubbles out into the
coolant stream must be separated and recycled. The effect
of dissolved gas on burnout is not yet known but may be
undesirable. One further limitation is that downflow must be
used and the liner must be strong enough to sustain the
pressure drop along the coolant channel. Although no work
is being done on this design at present, it may be examined
again within a few months.

(iv)   External Insulation

This is similar to the NPD and CANDU designs but with
the organic coolant there is a major problem in the coolant
tube material. However, as indicated previously, there is
promise that a zirconium alloy such as Zr-1.5 Al 10.5 Mo
either with or without a bonded aluminum liner may be
suitable. This is the most promising design from an economic
viewpoint.

SUPPORTING SERVICES

All the above programs require extensive analytical chemistry and
metallography support. In addition to carrying out routine determinations new
techniques have to be developed. In this we work closely with the USAEC and
to an increasing extent with Euratom.

Another major supporting service is in the operation of the loops at Chalk
River and the heat transfer and testing rigs at Peterborough.

CONCLUSION

One major problem with all organic systems is fouling of the fuel elements.
This has shown marked improvement in recent experiments and there is promise
that it can soon be reduced to an acceptable limit. Materials, another major
problem, is showing improvement in two directions; in the immediate future
there is prospect of better and more reproducible SAP products, and there
is promise that a successful zirconium alloy can be developed. This latter
material will have a real economic advantage. Many other problems are still
not solved but they are not major roadblocks to the development of an economic
OCDR.

TABLE 5

Engineers and Scientists on Canada's Organic Reactor Program

Project	AECL, Chalk River	CGE, Peterborough	Orenda, Toronto	Canadian Westinghouse, Hamilton	Total
WR-I Design	0	20	0	0	20
OCDR Optimization Studies	2	2	0	0	4
OCDR Research & Development					
- coolant decomposition	5	0	0	0	5
- fouling	3	0	0	0	3
- heat transfer	1	5	0	0	6
- materials & corrosion	1	4	1	3	9
- fuels	4	8	0	0	12
- coolant tubes	1	3	1	0	5
Supporting Services					
- laboratory	4	1	0	0	5
- loop and rigs	1	2	0	0	3
Totals:	22	45	2	3	72

Note: The current manpower shown fluctuates somewhat but these numbers apply for the next 6 months at least, with the exception of WR-I design effort which will double by the end of 1962.

For each professional there are 1 - 2 technicians and other assistants.

Organic Cooled Reactor Program in the United States

by

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The United States Atomic Energy Commission Organic Cooled Reactor Program
includes a broad effort on research and development and on operation of
experimental reactors, all of which would lead to the construction and
operation of nuclear power reactors of this type. The end objectives of
the program are to develop the technology necessary to the design, construc-
tion, and operation of economically competitive organic cooled power
reactors. The research and development is being carried out by Atomics
International at Canoga Park, California and the AEC's National Reactor
Testing Station in Idaho, Phillips Petroleum Company at Bartlesville,
Oklahoma and the AEC's National Reactor Testing Station in Idaho; Monsanto
Chemical Company at Everett, Massachusetts and Dayton, Ohio; California
Research Corporation at Richmond, California; General Electric Company at
the National Reactor Testing Station and Schenectady, New York; Massachusetts
Institute of Technology at Cambridge, Massachusetts; University of Utah;
University of California; Esso Engineering and Research Co. at Linden,
New Jersey, Aeroprojects Incorporated, at West Chester, Pennsylvania and
AEC's Brookhaven National Laboratory in New York. This coverage includes
work in coolant technology, fuels and materials, heat transfer and process
development.

The general effort by the above contractors and some recent significant results ᶜis outlined below:

I. Atomics International

    A. Coolant Technology

        1. A very important task related to the fuel element surface fouling problem is the determination of the significant physical properties of the Organic Moderated Reactor Experiment (OMRE)organic coolant (Santowax otho, meta, and para) and the effects of reactor operation, the decomposition products formed, and the role of oxygen (from air and water) on fuel element surface film formation

        2. Basic work is underway on the mechanisms of radiation damage to organic coolants using alpha radiation (polonium-210). This includes the yield of High Boiler (HB) and HB fractions from individuals irradiations of otho, meta, and para terphenyl and biphenyl.

        3. Work is underway to determine the inventory and distribution of radioactive materials in the OMRE primary system which will provide an indication of the relative hazards from coolant impurity activities and corrosion product activities.

        4. Studies are underway on the irradiation effects (fast neutrons vs. gammas) on organic coolants in order to determine quantitative effects.

        5. A small effort is underway on new and cheap organic coolants. The combined thermal and radiolytic stability of these new coolants in the 700-900°F range is being determined.

        6. The effect of high temperature (up to 900 F) on the thermal

and radiolytic damage of polyphenyl is underway to provide data for higher bulk coolant temperature.

7. Recently work has been initiated on the development of a chemical cleaning procedure for irradiated fuel elements, heat transfer surfaces, and the entire reactor piping system.

8. There is currently under intensive study the determination of the nature of the particles in the organic coolant, the method of formation of the particles, and the organic halo present in the particles. In addition, the relationship of this coating to the types of compounds present in the coolant, and the chemical changes which may take place during particle film formation are under study.

B. Heat Transfer and Process Development

1. Tests are underway on the determination of boiling heat transfer characteristics and critical heat flux as a function of operating parameters for organic coolants. Of particular interest is the effect of diphenyl and low boilers.

2. Bench-scale heat transfer studies are being conducted to study the many variables that may affect the heat transfer and fouling characteristics of various organic coolants.

3. A test has been initiated on the effects of fouling and dissolved gases on boiling heat transfer.

4. An intensive study is underway on means of impurity removal from organic coolant. Of particular interest is precoat filtration and fixed bed adsorption.

5. Coolant management studies are underway to lower make-up costs, to improve coolant characteristics and for over-all system simplification. Distillation studies are being conducted for high molecular weight high boiler separation.

6. In addition, there is coolant particulate removal equipment under development. There is work underway on precoat filters, adsorber columns, stainless steel, and glass filters, etc.

C. Advanced Engineering

This study investigates and evaluates advanced reactor engineering aspects of the various organic cooled reactor concepts that show promise of reducing power costs. The technical and economic feasibility of alternate moderators, fuel materials and new coolants is also being studied.

D. Fuel and Materials Development

1. There is currently underway corrosion and compatibility tests under static and dynamic conditions for various structural, cladding and fuel materials in polyphenyls.

2. An intensive effort is underway on the development of fabrication processes and in-pile testing of Aluminum Powder Metallurgy product (APM) clad UO_2 fuel elements which have high temperature and high burnup capabilities. The in-pile testing is underway in the OMRE.

3. There is also underway a modest program on the development of fabrication processes for Al clad, Nb bonded, U-alloy fuel elements. New fuel element shapes, fuel alloys and cladding material are being investigated and irradiation tests are to be conducted in OMRE.

4. In conjunction with the above, there is necessary pre- and post-irradiation inspection of OMRE stainless steel clad stainless steel UO_2 cermet driver, test elements of the U alloy type and UO_2 clad in APM.

II. Phillips Petroleum Company

1. **Basic Mechanisms** - There is underway a study of the means by which relatively stable aromatic molecules undergo radiolysis at high temperatures, the nature of decomposition products formed, and a correlation of the structural features which may have resistance to breakdown. Dosimetry studies under the high intensity Linac electron accelerator have received study using biphenyl and the terphenyl isomers. Information on the molecular weight distribution and the structure of high boiler was obtained.

2. **Film Formation** - Studies are essential to ascertain the mechanism by which polymeric materials build up on radiolysis of the coolant, the role of inorganic nuclei in catalyzing or agglomerating these materials, the means by which they migrate to the fuel plates and deposit thereon, and lastly the growth of heavy carbonaceous deposits. The purpose of the study is to find a means to break the film formation cycle. Mechanical removal and prevention of deposition by use of ashless detergent additives are being studied. Film deposition in a Linac circulating loop has been established and the effect of coolant velocity determined. The deleterious effects of iron particulate has been demonstrated as well as certain fractions of high boiler. The effect of hydrogen pressure in reducing film formation has been demonstrated. Studies on the role of oxygen contamination, carbidization and water content

were initiated, as were tests to determine electrostatic and short range irradiation effects.

3. New Coolants - Cheap industrial organics, particularly petroleum fractions, will be studied in a search for materials containing high percentages of polyphenyls or condensed ring aromatics and which demonstrate good radiation from thermal stability. Recent screening tests were performed on petroleum and coal tar oils and several promising fractions were found. Synthesis and screening irradiation tests were started on several cyclic nitrogen and aromatic types. The possibility of using benzane as makeup looks attractive to accomplish a considerable savings in make-up costs.

4. Reclamation - A significant reduction in make-up costs can also be effected if an appreciable fraction of the HB bottoms can be reclaimed as acceptable coolant. Investigation on hydro-cracking or other catalytic processes are being investigated.

5. Stabilizers - In order to stabilize and improve the properties of organic coolants, a program is necessary to understand the mechanisms by which additives may render them more suitable. Industrial additives for industrial oils may be feasible. A study of materials which will control the increase of viscosity upon radiation is also of interest.

III. Monsanto Chemical Co.

1. Thermodynamic and Physical Data - The use of organic working fluids in direct cycle turbine applications is of long range interest. Data are being investigated in order to be able to select an optimum coolant and establish the performance of this concept.

2. __Reclamation__ - Hydrocracking catalysts have been developed. Reclamation by solvent distribution was demonstrated to be technically and economically feasible.

IV. __Massachusetts Institute of Technology__

In 1961 the MIT in-pile loop was completed, tested and Santowax OMP was irradiated. This program draws upon the coolant R&D at other sites and feeds back irradiation data on their selected materials and processes.

V. __University of Utah__

The need exists for theoretical studies on the effects of temperature and radiation on bond stability for carbon-crabon single, carbon-carbon double, carbon-carbon triple, carbon-carbon resonating and carbon-hydrogen bonds. In 1961 structure theory was developed for biphenyl and an effort was initiated on producing reactions of ferric iron and polyphenyls.

VI. __California Research Corporation__

1. __Film Formation and Polyphenyl Decomposition__ - There is underway hot wire and electron radiation tests to study film formation. The Shield Pool Test Facility at NRTS was used to conduct fast neutron and gamma radiation experiments.

2. __New Coolants__ - Promising petroleum stocks have undergone thermal capsule tests and irradiations in the Materials Testing Reactor shuttle tube.

VII. __University of California__

Studies will be performed to obtain data on Pressure, Volume, Temperature relationships for biphenyl, otho, meta, and para terphenyl

and other organics from room temperature up to the critical temperature.

VIII. <u>Brookhaven National Laboratory (BNL)</u>

The BNL has recently been established as the cognizant AEC Laboratory in organic cooled reactor program. They have a broad program underway which will provide an understanding of: (1) The mechanisms of fouling film formation, (2) corrosion of metals, and (3) the stability of condensed aromatic and polyphenyl compounds as functions of carbon-to-hydrogen ratio, resonance energy and molecular structure. The program will be expanded to include coolant purification, the effect of fission product release into the coolant, fluidized fuel reactor cores, direct-cycle organic reactors, and related problems.

IX. <u>General Electric at General Engineering Laboratory</u>

It is planned to determine whether an organic moderated reactor could be operated continuously without clean-up or replenishment of the organic fluid by utilizing an electrochemical device in an external loop by undertaking an experimental program to study the nature and characteristics of steady state in organic liquid under soft corona discharge. Should this prove feasible a substantial decrease in the cost of coolant makeup could be realized.

X. <u>Esso Engineering and Research Co.</u>

This is a new program which is aimed at identifying , by $Carbon_{14}$ labeling, the extent to which various constituents in irradiated organic moderator coolants contributes to the fouling of fuel element surfaces. Studies will also be undertaken on the development of procedures for

decontaminating irradiated organic moderator coolants, particularly those which contain sulfur35 and phosphorus32. This is of interest as sulfur and phosphorus compounds are effective inhibitors for coolant degradation.

XI. Aeroprojects, Incorporated

This is a recently initiated program on the possible use of ultrasonics for removal of fuel element surface film deposits for re-use in the reactor for chemical reprocessing. This will be demonstrated on test coupons, in the hot cell on irradiated OMRE fuel, and finally at OMRE on fuel that could be reinserted into the core. Initial results on simulated films indicate that this may be feasible.

XII. Other

In addition to the above research and development program, we are operating or constructing experimental reactors such as the OMRE and the Experimental Organic Cooled Reactor (EOCR). The OMRE is the original reactor experiment and, since its early success, has been used as an experimental facility for testing coolants, fuel elements, and process systems. The reactor is currently operating on Core III. The initial part of Core III operation, designated as Core IIIA, began in July and terminated in October, 1961 with an accumulative exposure of 282 MWD. Examination of driver fuel elements (UO_2-SS cermet clad in SS) indicates that the operation was successful as the average film thickness was less than .1 mil. The coolant purity was maintained at about 10 ppm. in organic matter by the use of continuous distillation.

The continued operation of core III (designated core IIIB) will include some of the driver fuel elements from IIIA, two aluminum clad uranium-molybdenum alloy fuel elements, and four APM clad UO_2 fuel elements. One of the APM-UO_2 fuel elements was in Core IIIA and has successfully operated to a burnup of about 2000 MWD/MT. These fuel elements will be tested to 25-30,000 MWD/MT and at surface temperatures up to $850^{\circ}F$. The core IIIB will be operated with cleaner coolant than IIIA by increasing the distillation rate and using filters. Current plans for Core IV are to replace all the driver fuel elements with Al clad U-Mo type fuel elements and to continue to irradiate the APM-UO_2 test elements.

The EOCR (20-40 MWt) is about 85% complete and will be operational by mid 1963. The reactor will be completed with two large in-pile test loops. The fuel technology loop will be $6\frac{1}{2}$ inches in diameter and capable of generating $1\frac{1}{2}$ MW of thermal energy from a large fuel test specimen. The coolant technology loop is $2\frac{1}{2}$ inches in diameter and will have a high in-pile to out-of-pile volume as it will make two passes through the core. The initial operating core will be made up of SS clad UO_2-SS cermet drive fuel elements. There are also provisions for test elements in the periphery of the core. The initial test program in the EOCR will be concentrated on the current coolant and types of fuel elements. At a later date improved coolants, higher temperatures, and improved fuel elements will be developed in the loops and the core positions.

It is expected that the City of Piqua Power Reactor (11.4 MWE) will be critical in July and at full power by the end of this year. The maximum fuel element surface temperature is 713°F with an average coolant outlet temperature of about 575°F. The expected average burnup is 3,000 MWD/MT. This reactor will demonstrate the feasibility of a small organic cooled and moderated power reactor.

In order to advance the technology toward competitive nuclear power, an advanced type of reactor has been under development. The project that would demonstrate this technology is the Prototype Organic Power Reactor (POPR). This project has been indefinitely deferred and the supporting R&D program curtailed except for the APM-UO$_2$ fuel element development. However, very recently an expression of interest in regard to an Organic Power and Heat Industrial Reactor (OPHIR) was submitted to the AEC for a cooperative arrangement with the Packaging Corporation of America (PCA) at their Filer City, Michigan paper and pulp mill. PCA would provide a site, operate the plant, and utilize the steam for both process heat and electrical power production using their turbine-generators. The nuclear steam supply system is essentially the same as the POPR and will demonstrate both the POPR objectives and industrial process heat applications of nuclear power. The AEC is currently awaiting a formal proposal for the project. The organic process heat application is particularly attractive because of the low pressure primary system which would minimize the possibility of contaminating the process steam due to steam generator leaks. The

OPHIR would demonstrate and prototype for a 300 MWE plant many advances over the Piqua reactor such as APM-UO$_2$ fuel, higher temperatures (surface temperature, 850°F-coolant outlet, 700°F), in vessel fuel handling, hydraulic control rod drives, improved coolant purity control, and failed fuel element detection system.

There is also under evaluation the heavy water moderated organic cooled natural uranium power reactor. This combination has many inherent advantages and the potential problems with such a system will be examined. A cost optimization study of a 300 MWE plant and substitution physics experiments using metal and oxide fuel are underway at duPont. In addition, United Nuclear Corp. is to perform substitution experiments using uranium carbide fuel. We have recently added to this a design study of a 500 MWE plant which would be able to operate on natural uranium up to 5,000 MWD/MT but be optimized for enrichment to minimize the fuel cycle costs. In conclusion, our trend at the current time is to examine the organic moderated and the heavy water moderated power reactors while continuing to support the basic research and development and the OMRE operation. In the near future, the Piqua reactor and the EOCR will be making major contributions toward this aim.

Organic Cooled Reactor Program Budget

(millions)

Activity	FY 1961 (Man Years		FY 1962 (Man Years)		FY 1963 (Man Years)	
	$	MY	$	MY	$	MY
(1) OMRE - Includes analysis and evaluation, maintenance and operational development, fuel fabrication of cores and operation.	2.0	60	~2.1	60	~2.1	60
(2) EOCR - Includes R&D operation and fuel fabrication	.8	25	~1.3	44	~2.3	65
(3) Other Research and Development	~3.9	80	~4.3	100	~4.4	110
(4) POPR Research and Development	1.0	38	~1.6	44	--	-
(5) Piqua - Fuel fabrication and operations	1.0	-	.9	-	.5	-

ORGANIC COOLED REACTOR CHARACTERISTICS AND SCHEDULES

	TMW	MWE	Fuel and Enrichment (%)	Coolant Temperature ($^{\circ}$F)		Hot Spot,$^{\circ}$F Temperature
				Inlet	Exit	
OMRE	12	-	SS-UO$_2$-F.E.	575	600	750
Piqua	45.5	11.4	U-Mo 3.5 - 1.94	518	575	715
EOCR	40 or 20	-	SS-UO$_2$- F.E.	500	525	850
POPE	160	50	APM-UO$_2$- 3.0	575	700	850

	Status of Const.	Const. Comp. Date	Criticality Date	Power Date	Plant Cost
OMRE	Compl.	Compl.	1957	1957	2.4
Piqua	100	Jan. 1962	July 1962	Sept 1962	8.2 M[1]
EOCR	~ 86	July '62	Feb. '63	July '63	9.8 M (excls. 2 loops at 3.7 M

F.E.   Fully Enriched

(1)   Nuclear Steam Supply System

Heat Transfer and Burnout Studies

by

R. C. Johnston

Canadian General Electric Company Limited

Heat Transfer and Burnout Studies

1. ## Introduction

The short range Canadian program of heat transfer and burnout studies may be divided into three areas:

(a) Dynamic studies of ONB and DNB heat fluxes using simple geometries.

(b) Coolant screening studies to investigate the effects of impurities and coolant composition on ONB and DNB heat fluxes.

(c) Dynamic studies of ONB and DNB heat fluxes using multiple-rod geometries.

The objectives of this short range program are:

(a) To develop the necessary information to establish design limits and operating conditions on the first generation fuel for large organic cooled power reactors (OCDR).

(b) To establish optimum or limiting coolant conditions insofar as composition and impurities affect heat transfer.

(c) To verify the extrapolated heat transfer data being used in the design of the Organic Test Reactor (OTR) which will be located at the Whiteshell Nuclear Research Establishment.

To date one coolant has been considered for the OCDR and OTR designs. This is Monsanto's Santowax OM with radiolytic high boilers and having the following composition range.

Reference Coolant

Component	Weight %
Biphenyl	3 maximum
Ortho-terphenyl	40 - 50
Meta-terphenyl	15 - 20
Para-terphenyl	1 - 2
Radiolytic high boilers	30 nominal

This is the reference coolant for all the heat transfer studies. The first generation fuel for both reactors is similar in that each fuel rod will consist of bundles of cylindrical elements. Therefore, all the work in the present heat transfer program has employed cylindrical heated sections to simulate the fuel elements.

At present there are no detailed plans to extend the work to other geometries or for using other coolants.

2. Dynamic Heat Transfer Studies - Single Geometries

2.1 DNB and Burn Out Heat Fluxes

This work has been done using an annular flow test section with a direct heated inner tube of 0.500 inches OD (1.27 cms) and a heated length of 14 inches (35.6 cms). A balanced bridge type of burnout protector was used to terminate a number of the runs short of physical burnout. Comparison of runs using the burnout protector and runs which proceeded to physical burnout showed the critical heat fluxes obtained using the burnout protector were within 5% of the heat fluxes obtained by physical burnout.

It is planned that burnout data be obtained for the variables over
the following ranges:

Velocity	5 - 35 ft/sec	(1.5 - 10.7 m/sec)
Bulk temperature	700 - 800°F	(370 - 425°C)
Sub-cooling	50 - 300°F	(30 - 170°C)

Table 1 is a summary of the pertinent burnout data that has been
obtained. It will be noted these runs generally cover the conditions of
interest to OTR. The continued work will complete the range filling in the
gaps where necessary.

The data of Table 1 are plotted in Figure 1 using a Core and Sato
type plot. The plot shows the general agreement between values obtained from
physical burnout and those obtained when the run was terminated by the burnout
protector. For comparison the Core and Sato line for Santowax OMP with its ±
2 σ limits is also shown on the plot.

Although the experimental program and the analysis and interpretation
of the existing results are not complete it is apparent that the DNB heat
fluxes for the reference coolant, which contains about 30% high boilers, are
significantly higher than the Core and Sato data for both Santowax R and
Santowax OMP. The present results are in good agreement with recent AI data
obtained using Santowax R with 24% high boilers (i.e. about 35% high boilers
in all).

Current CGE fuel design is based on the use of a factor of safety of
two on the Core and Sato OMP - 2 σ line as far as the DNB heat flux is
concerned.

Although the foregoing results indicate that this criterion is quite
conservative, further information is deemed necessary before the maximum
permissible design heat fluxes can be raised. The objectives of the ONB

studies and the multiple-rod geometry studies described in sections 2.2 and 4 of this paper are to obtain this information.

2.2 Onset of Nucleate Boiling Studies

One of the objectives of the current program is to obtain data on the heat flux at the onset of nucleate boiling with the reference coolant. Data have been gathered in this area but so far no great amount of reduction or interpretation has been done. However, some preliminary statements can be made.

Because of the presence of coolant components of different volatilities and of dissolved gas it is difficult to define the exact location of the onset of nucleate boiling. This is illustrated in figure 2 which shows a typical plot of heat flux versus overall temperature difference obtained under dynamic conditions. Further experiments plus interpretation of existing data are proceeding on this aspect of the program. Preliminary results have indicated, however, that variation in the low-boiler fraction of the coolant between about 0.2 to 1.5% has a very significant effect on the heat flux and corresponding overall temperature difference at which experimental points begin to depart from the forced convection line.

Part of the ONB Investigation is directed towards obtaining friction factors for conditions beyond the forced-convection region in order to assess the implications of reactor operation in this region from the point of view of pressure drop and flow stability.

3. Coolant Screening

The ONB and DNB work mentioned in 2 is being carried out using the reference coolant and with the gas content within a range of about 200 to 700 ml/kg. The effects of impurities and coolant composition on ONB and DNB are being investigated separately using a pool boiler. If the

pool boiler work shows a significant effect due to any one variable then that variable will be investigated further under dynamic conditions in SOTR.

The variables being covered in the present phase of this work are:

Water	50 - 600 ppm
Low boilers	up to 2% by weight
High boilers	up to 40% by weight

The effect of gas content is being covered separately by work now in progress at A.I.

The first variable being investigated is water. Runs have been made with water content varied from about 60 ppm to 630 ppm. Preliminary conclusions from this work are that, within the limits of experimental accuracy the presence of dissolved water in the reference coolant has no effect on the DNB heat flux in pool boiling for dissolved water contents up to 630 ppm. Figure 3 illustrates this.

4. Dynamic Studies of ONB and DNB - Multiple Rod Geometries

The objective of this work is to determine the applicability of data obtained in simple annular geometries to complex geometries having non-uniform velocity distributions and multiple-heated surfaces. The work will be done in HTOTR using a test section consisting of two directly heated parallel tubes, simulating fuel elements contained in a flow tube which will be shaped to simulate adjacent fuel elements. Coolant composition, velocity, sub-cooling and bulk temperature will be representative of the range covered in the annular geometry studies.

Since coolant hold-up in the reactor core adversely affects reactor economics, the study is intended to establish the minimum acceptable

element to element spacing. The initial spacings to be investigated are in the range of 0.020 inches to 0.060 inches. The first test will be run with element spacing of 0.040 inches (1 mm), the element spacing in the current OTR reference design. Whether the next succeeding test is run with a larger or smaller spacing will depend on the results of the first test. Selection of the spacing for the third test will be influenced by the data obtained on the earlier tests.

Fabrication of the first test section is now complete and it is being installed in the rig. Testing will begin about the middle of June.

5. Other Heat Transfer Studies

To gain an understanding of the fundamental effects of coolant components of different volatilities on the DNB heat flux under dynamic conditions AECL is conducting experiments with toluene and benzene. This work is at an early stage and no results are available to report.

6. Vapour Pressure Determinations

In all of the preceeding work the understanding and correlation of data is dependent on an accurate knowledge of the coolant vapor pressure. Progressive refinements have been made in the measuring techniques and it is now felt that measured values of vapour pressure are within $\pm$ 5%.

Table I: Preliminary Single Element DNB Results

Run	P Psia	T_B °F	V ft/sec	ΔT_{sub}* °F	$\Delta T_{sub} \, V^{2/3}$	Q/A (Btu $Hr^{-1} ft^{-2}$) Trip	Burnout
SOTR							
4	76.3	700	10.1	127	595	561,000	
6	73	692	10.1	131	613		592,000
12	99	726	21.6	151	1,180	827,000	
14	101	706	21.6	174	1,360	825,000	
17	206	724	13.9	276	1,600		1,280,000
19	98	714	5.3	162	493		526,000
20	99	701	31.8	177	1,770		1,268,000
21	151	703	11.5	241	1,230		985,000
22	163	701	10.5	256	1,230		983,000
23	202	698	10.8	298	1,460		790,000
32	99	703	10.0	175	813	695,000	
HTOTR							
9	97	761	10.0	109	505		585,000
11	182	704	9.9	270	1,240		1,140,000
12	88	699	10.1	156	730		612,000
14	148	671	4.7	265	747		640,000
15	201	696	10.2	298	1,400		706,000

* - Note that subcoolings are based on saturation temperatures calculated from coolant compositions on xylene-free basis. The above results were obtained with coolants having four somewhat different compositions.
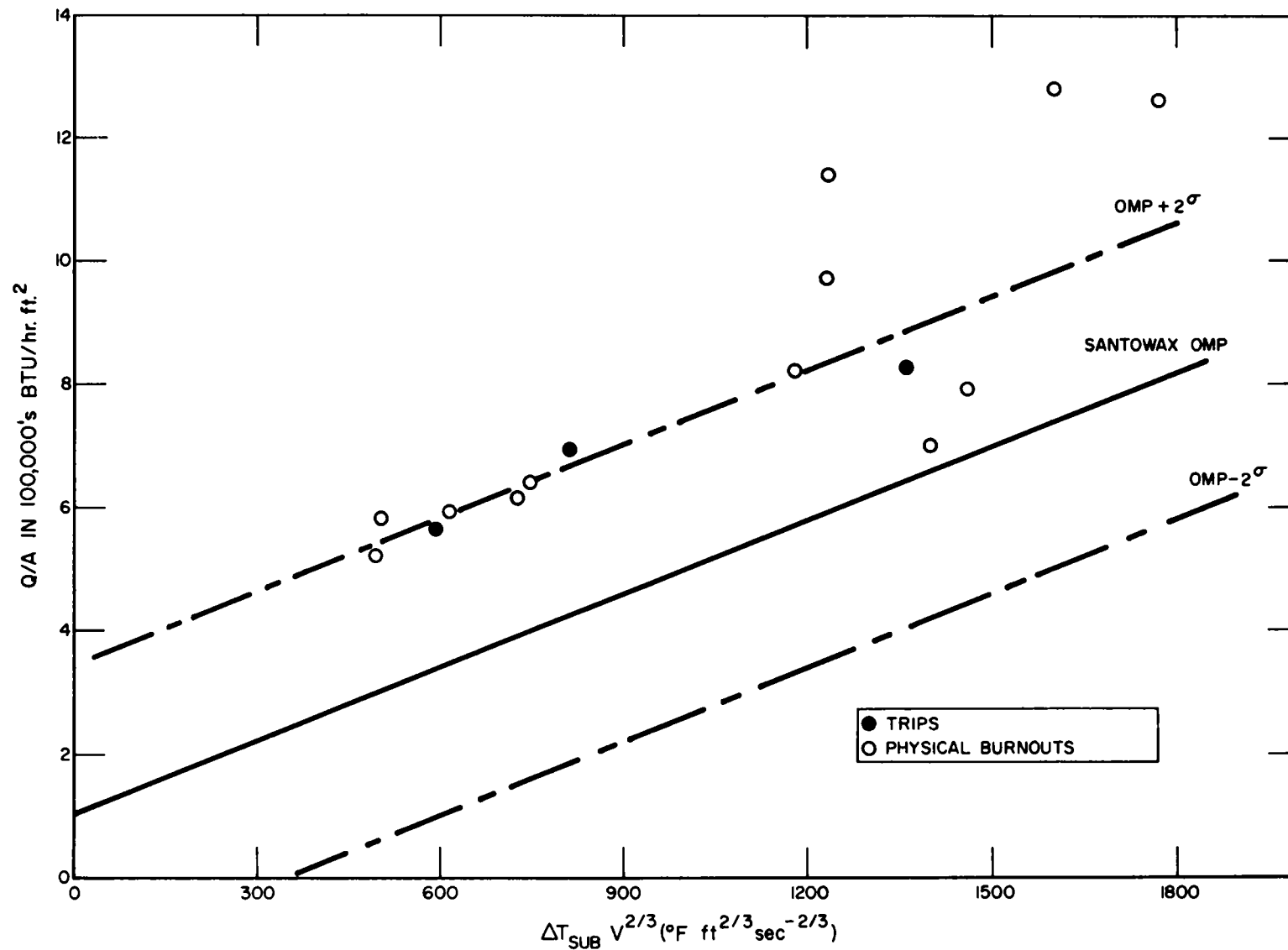
Figure 1. Preliminary Single Element D. N. D. Results on Xylene — Free Basis

Figure 2. Typical Boiling Curve
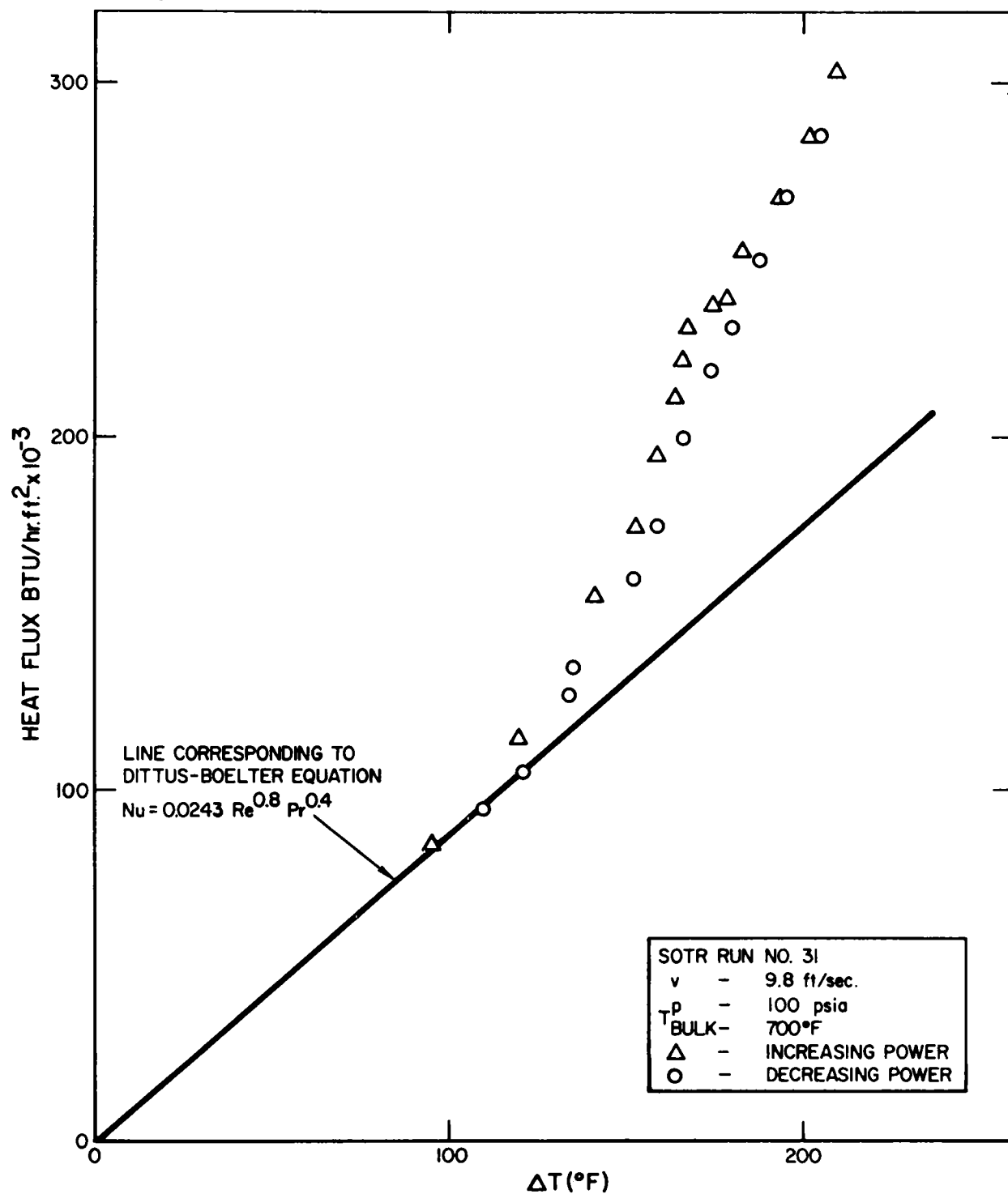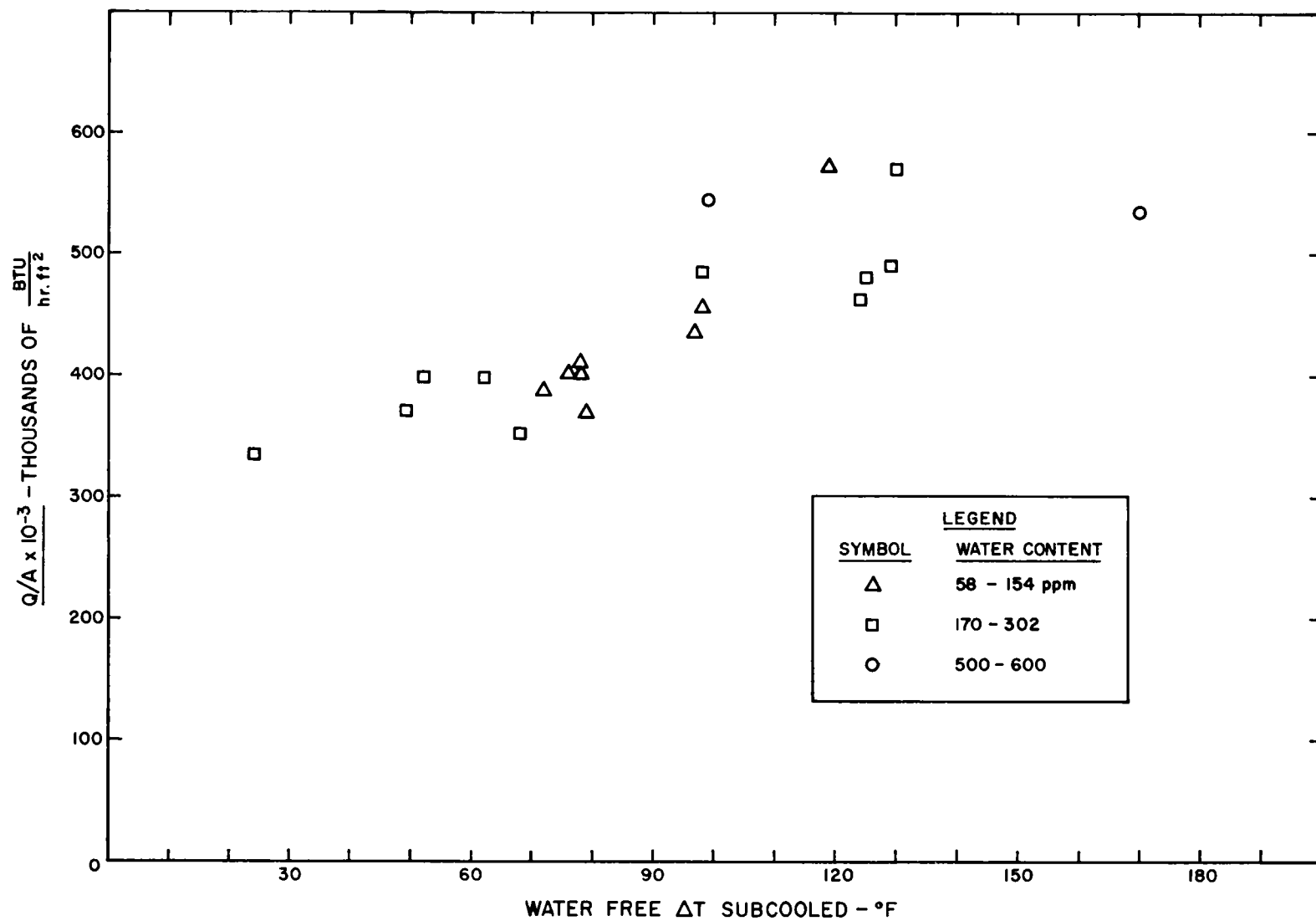
Figure 3. Pool Boiler DNB Heat Fluxes (Using organic coolant with varying low water contents. Subcoolings calculated using water free saturation temperatures)

OCDR

COOLANT TUBE ASSEMBLY DEVELOPMENT

by

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1.0 <u>INTRODUCTION</u>

During the past year the emphasis in our coolant
tube development program has changed. Previously, work was
concentrated on the development of a readily workable fuel
channel which would give acceptable power costs in a small
prototype power reactor. Now, however, the emphasis is on
the development of a few coolant tube concepts that offer the
greatest economic potential in a large power reactor.

2.0 <u>COOLANT TUBE ECONOMICS STUDIES</u>

To determine which of the many concepts under con-
sideration offered the greatest potential, computer optimiza-
tion studies were carried out and as a result, we are con-
tinuing the development of only four concepts and placing em-
phasis on the development and use of more economical materials.

Each coolant tube concept was optimized for an organic cooled, natural uranium fuelled and heavy water moderated power reactor with a gross output of 200 eMW. The results of this optimization are shown in Fig. 1 - "OCDR Fuel Channel Economic Assessment". This plot shows the total unit energy cost in mills/kw.hr. VS $\sum(\Sigma_a Rt) \times 10^{-3}$. Each fuel channel is divided into a number of homogeneous annular regions where Σ_a is the macroscopic absorption cross section of each region in cm^{-1}, "R" is the mean radius of the region in cm and "t" is the thickness of the region in cm.

As a result of these optimizations, our development work has been limited to the following fuel channel concepts or slight variations thereof:

1. Solid insulated fuel channels using zircaloy stress tubes, SAP liners and MinK insulation.

2. Organic insulated fuel channels using zircaloy stress tubes and SAP liners.

3. Gas insulated cold tubes.

4. Zircaloy hot stress tubes with zircaloy calandria tubes.

3.0 STATUS OF THE CURRENT DEVELOPMENT PROGRAM

Currently, only the solid and liquid insulated fuel channels are under active development. The gas insulated cold tube is being studied and the zircaloy hot stress tube awaits a satisfactory means of preventing hydrogen embrittlement of the zircaloy.

3.1 Solid Insulated Fuel Channel Development

The solid insulated fuel channel concept we are developing consists of a 0.76 mm (0.030 in.) wall zircaloy stress tube with a 1.52 mm (0.060 in.) wall SAP liner tube joined at each end, beyond the core boundaries, by sandwich type roll formed joints as reported by Brown[1]. Between the liner and stress tube a fibrous compressed silica insulation of 2.54 mm (0.100 in.) thickness known as MinK is fitted in segmented rings.

To date, four short models 1 meter (3 ft.) long and one prototype 3 meters (10 ft.) long have been built and tested employing various types of insulants. Each of the models and the prototype were thermally cycled from 40 to 370° C at 21 kg/cm^2 (100 to 700° F at 300 psig) - however, the model tests were carried out in a stagnant pool of petroleum oil so that metal temperatures were below the maximum oil temperatures. The prototype was tested at Canadian General Electric in MOTR at actual OCDR conditions.

A great deal of information has been accumulated from these tests, both the successes and the failures. The first model performed quite satisfactorily for 100 cycles. The second model failed due to the slippage of a rolled joint. The third model performed quite satisfactorily for 100 cycles after which the liner was purposely defected which caused it to fail (Fig. 2). The fourth model failed after 80 cycles due to a serious stress concentration in the liner. The prototype (Fig. 3) had completed 9 cycles when the liner failed.

The following general conclusions may be drawn from the results of the entire development program on solid insulated fuel channels.

1. The concept is attractive from the points of view of economics and heat loss. $H \approx 15 \; \frac{kg \; cal}{hr. \; m^2 \; {}^oC}$ (3 BTU/hr.ft.^{2} ^{o} F).

2. The inherent safety of the design is very good. In all failures to date, the zircaloy stress tube has not suffered any damage whatsoever.

3. The present design has proven acceptable from a manufacturing standpoint and the use of insulation in formed segments is feasible.

4. None of the insulations tried to date have suffered any damage or have posed any serious problems in the development work.

5. With the presently available SAP materials this concept is more marginal than anticipated, however all failures to date have occurred with organic in the insulation annulus or, in the case of the fourth model, with a serious stress riser in the liner.

3.2 Organic Insulated Fuel Channel Development

The organic insulated concept which is the subject of all of our recent development work is a narrow annulus between the liner and stress tube interrupted by a series of

widely spaced rings machined on the liner which form close fitting restrictions to the annulus flow.

A very detailed theoretical analysis and development program has been carried out on this concept by Judd[4,6] and Rogers[5,7] and the theoretical predictions have been very well substantiated by the development results.

In addition, a test section for X-7 in NRX is under construction employing a SAP liner and a zircaloy stress tube which is designed to obtain in-pile data on the organic insulant and the use of zircaloy in contact with organics.

From the work carried out to date, we have reached the following conclusions on the liquid insulated fuel channel.

1. Overall heat transfer coefficients of approximately 244 kg. cal/hr.m^2 $^\circ$C (50 BTU/hr.ft.^{2} $^\circ$F) can be achieved for an annulus thickness of 0.76 mm (0.030 in.) and for a pressure drop of 3 kg/cm^2 per meter (15 psi per ft.). Test results are shown in Fig. 4.

2. Larger ring pitches reduce the heat transfer but there is very little gain beyond a pitch of 20 cm (8 in.).

3. The effect of ring clearance is not nearly so well defined. Even with predicted hot interferences of 0.07 mm (0.030 in.) between the rings and the stress tube annulus flow has been maintained but this has been done with the simulated moderator temperatures close to boiling. When the water temperature is depressed to more realistic moderator temperatures –

i.e., 50° C (120° F), larger hot clearances are required
to maintain annulus flow. Tests are being carried out to
try and evaluate these effects more accurately.

4. For minimum heat transfer the ring width should be minimized.

5. The low insulating annulus Nusselt numbers achieved with
   the widely spaced ring configuration (as low as 1.80 at
   3 kg/cm^2 per meter (15 psi per ft.) pressure drop) indicate
   that not much absolute improvement could be expected with
   any other practical annulus configuration employing a
   flowing organic coolant.

3.3 Solid Insulation Evaluation

The most promising insulant investigated to date has
been MinK 3120, a product of Johns-Manville which is available
in preformed segments of 120° each and 15 cm (6 in.) long which
are fitted together to form continuous rings between the liner
and stress tubes. It is a fibrous material composed mainly of
silica with very low thermal conductivity 0.0335 $\frac{\text{kg cal}}{\text{hr. m } ^\circ\text{C}}$
(0.0225 $\frac{\text{BTU}}{\text{hr.ft.}^\circ\text{F}}$) and macroscopic cross section 0.0019 cm^{-1} and
is capable of supporting a load of 70 kg/cm^2 (1000 psi) with a
5% recoverable deformation.

To date, test experience with MinK has been limited
to the fourth model but its performance appears completely
satisfactory. Short term irradiation tests are now underway
in X-7 to look for changes in physical properties but these

can only be regarded as preliminary to long term in-pile irradiation of a complete fuel channel.

Fired ceramic rings such as zirconia are no longer being considered because of the difficulty in manufacture and consequent high cost, their relatively high thermal conductivity and macro cross section and doubts about their radiation and structural stability under reactor conditions.

3.4 Rolled Joint Development

Development work on rolled joints has been limited to those joints that were actually required to build models and prototypes. As a result of this work the "club sandwich" joint was developed which is of the broad interlocking groove type as shown in Fig. 5. This joint is used to join the SAP liner and the zircaloy stress tube to the steel extensions of the solid insulated fuel channel and it has proven to have adequate strength and leak tightness. Test results on a sample joint are also shown in Fig. 5.

4.0 FUTURE DEVELOPMENT TRENDS

There are several areas of our development program where emphasis will result in more economical and completely reliable fuel channels.

The first of these is in the improvement of material properties. SAP tubing we have been using has lower than anticipated physical properties and less than desirable uniform quality to meet the stringent requirements of the solid insula-

ted coolant tube. SAP tubing is now on order which will have physical properties adequate to permit the liner to operate within a stress range that is completely elastic.

A source has now been established for thin wall welded zircaloy tubing which is subsequently drawn to improve its physical properties which will be adequate for in-pile applications of solid insulated fuel channels.

Design innovation offers a second possible way of improving fuel channel designs. One method of reducing the axial thermal stresses in the solid insulated concept is to use a bellows in the liner or to use thermally compensating fuel channel extensions. Another approach to the solid insulated liner problem is to use a ductile aluminum liner which would have a relatively large cyclic life before failure occurred.

The third area where future development promises large rewards is in the use of materials of lower cross section. The obvious application here is the use of zircaloy in organics at coolant temperatures as described by Dr. W. M. Campbell.

The fourth area where considerable economic improvement is available is in the use of optimum coolant tube, coolant and fuel cross sectional areas.

Fig. 6 shows three methods of obtaining or approximating ideal conditions of minimum coolant area, uniform coolant velocity and minimum fuel channel material. The first of these shows the M-1 fuel channel suggested by Dr. A. J. Mooradian of

AECL which consists of seven individual hot stress tubes, each containing one fuel element, with one calandria tube surrounding the bundle. The second shows a solid insulated coolant tube with a fluted liner which displaces the large organic areas around the periphery of a seven element bundle with insulation and liner material. This fuel channel when optimized for a large power reactor reduced the unit power cost by 0.20 mils/kilowatt hour and increased burnup by 1150 MWD/TeU when compared to a similar circular solid insulated fuel channel.

Possibly the best way to achieve the optimum fuel-coolant-fuel channel relationship is with a circular fuel channel and annular fuel. This arrangement will provide the minimum coolant tube, the minimum coolant area with uniform coolant velocity and maximum fuel area for a given fuel channel.

This review of our development work is necessarily brief but more detailed information is available for discussion.
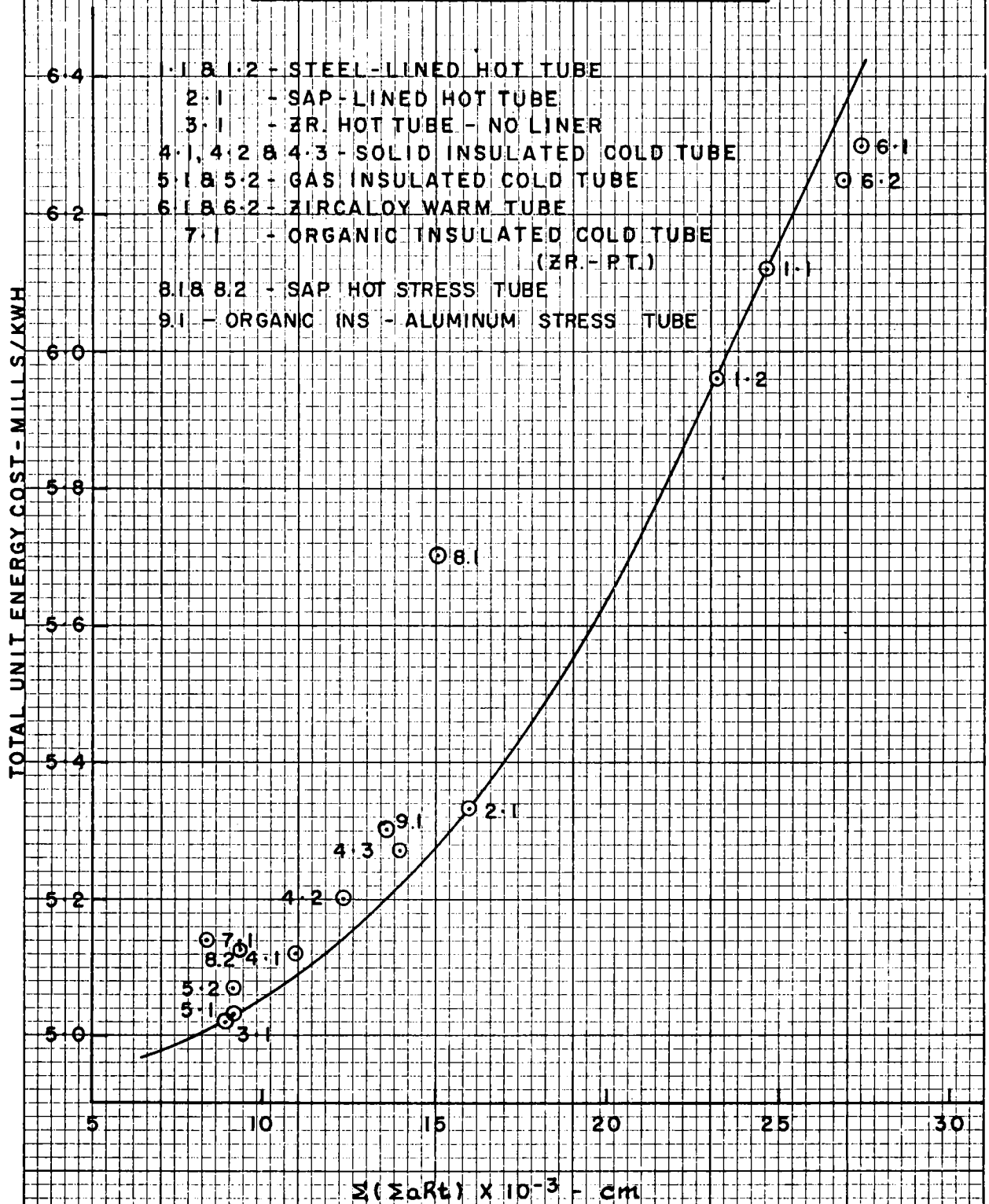
Figure 1 - OCDR Fuel Channel Economic Assessment
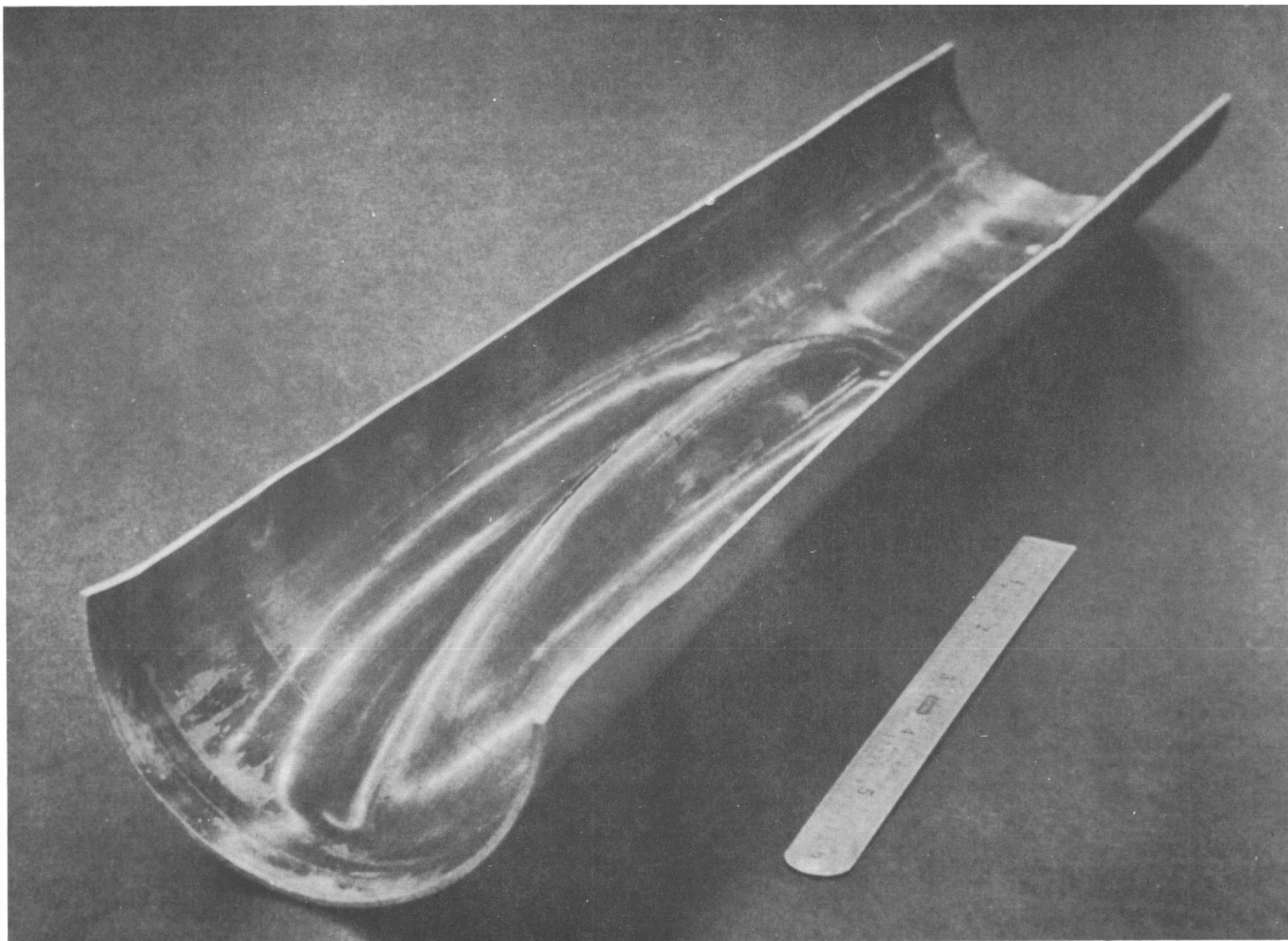
Figure 2 - Blister Formed on the Liner of the Third Model
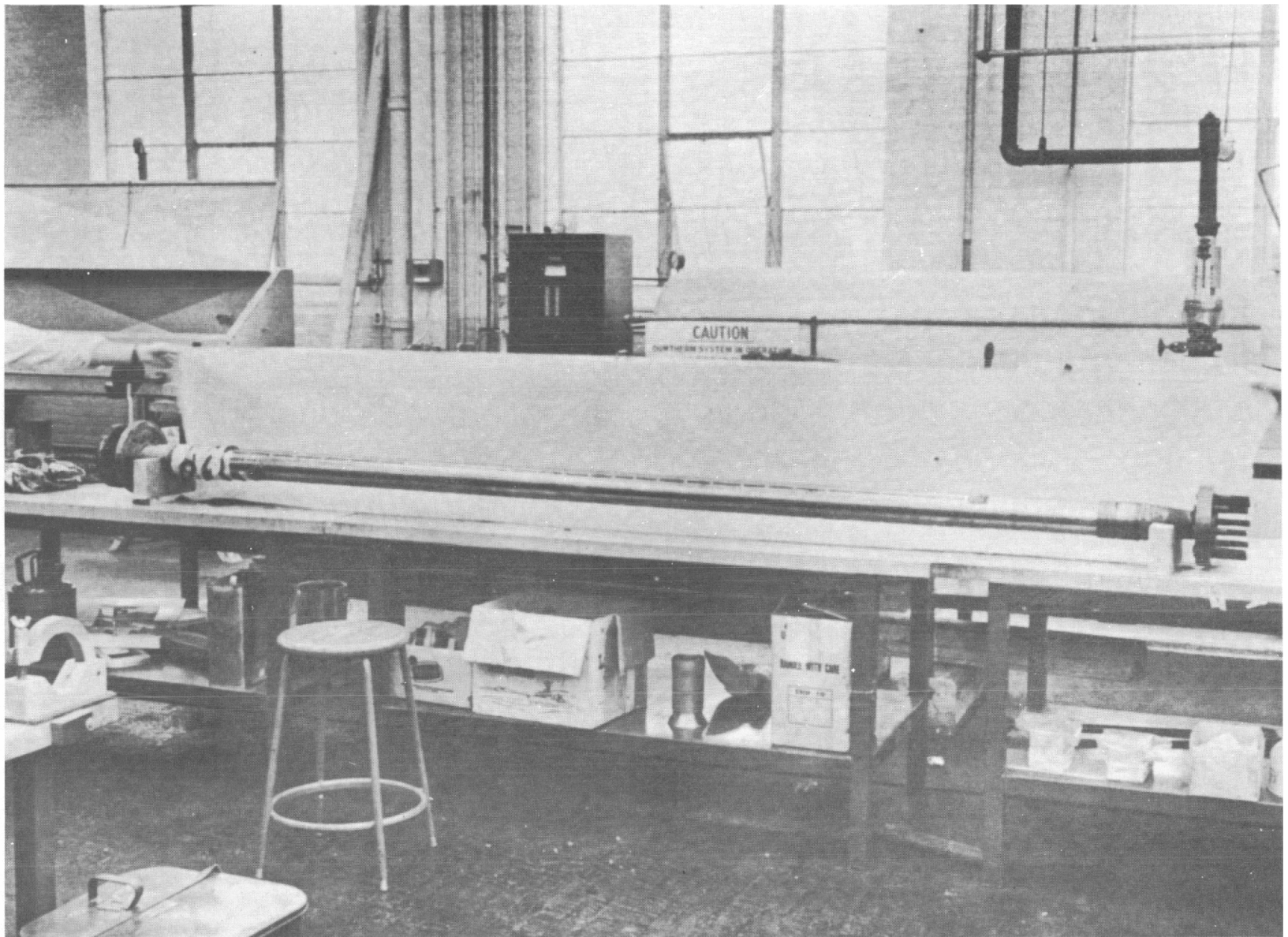
Figure 3 - First Prototype Solid Insulated Fuel Channel
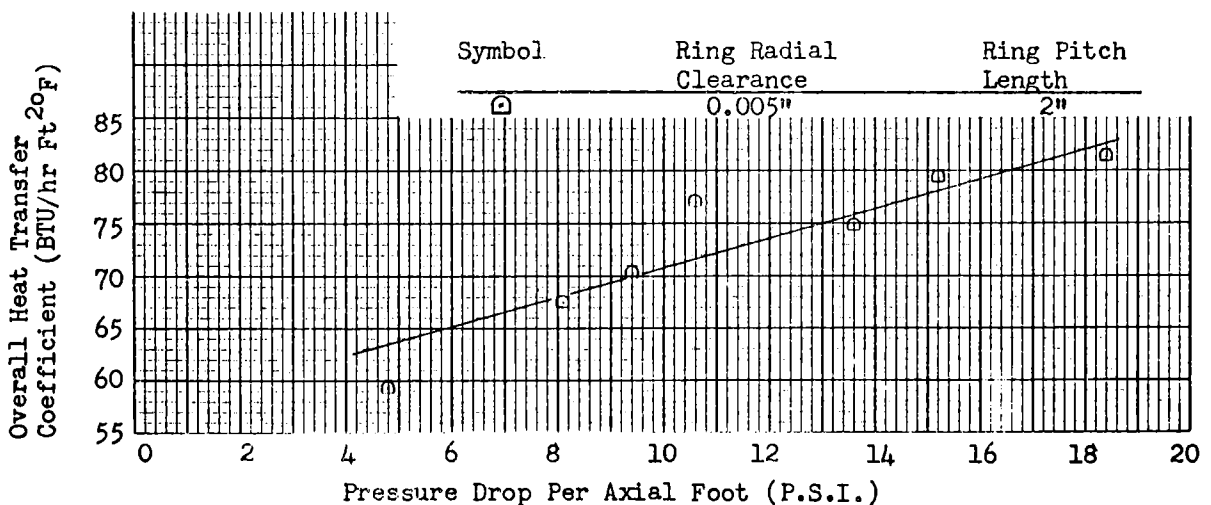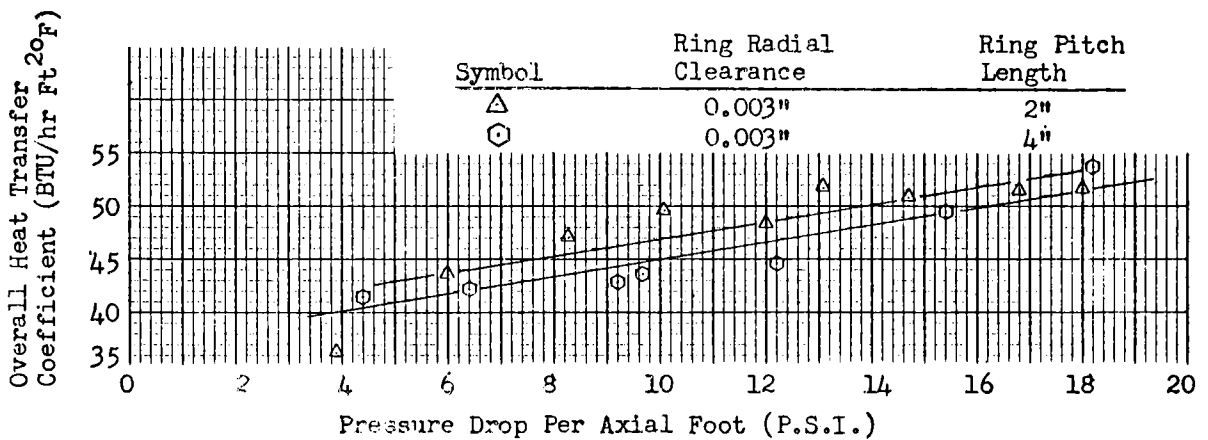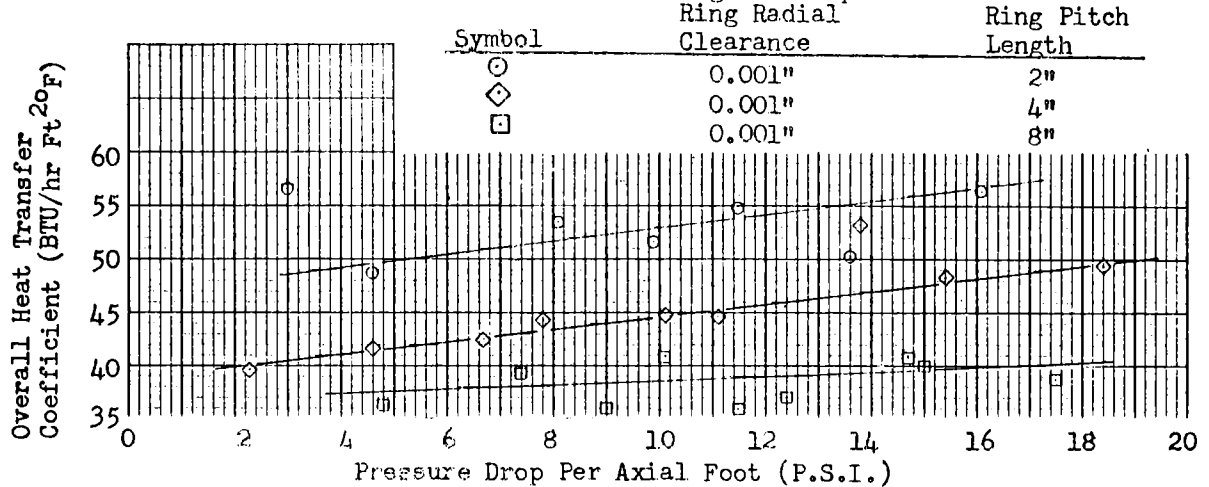
Figure 4 - Heat Transfer Coefficient vs. Pressure Drop

for the Organic Insulated Fuel Channel

BACK WELD
AFTER ROLLING

STN. STL.

2 3/8

.070
S.A.P.

.015 (TYP.)

.035
ZIRC.

3.660
DIA.

3.220
DIA.

7/16

.010

TEST RESULTS
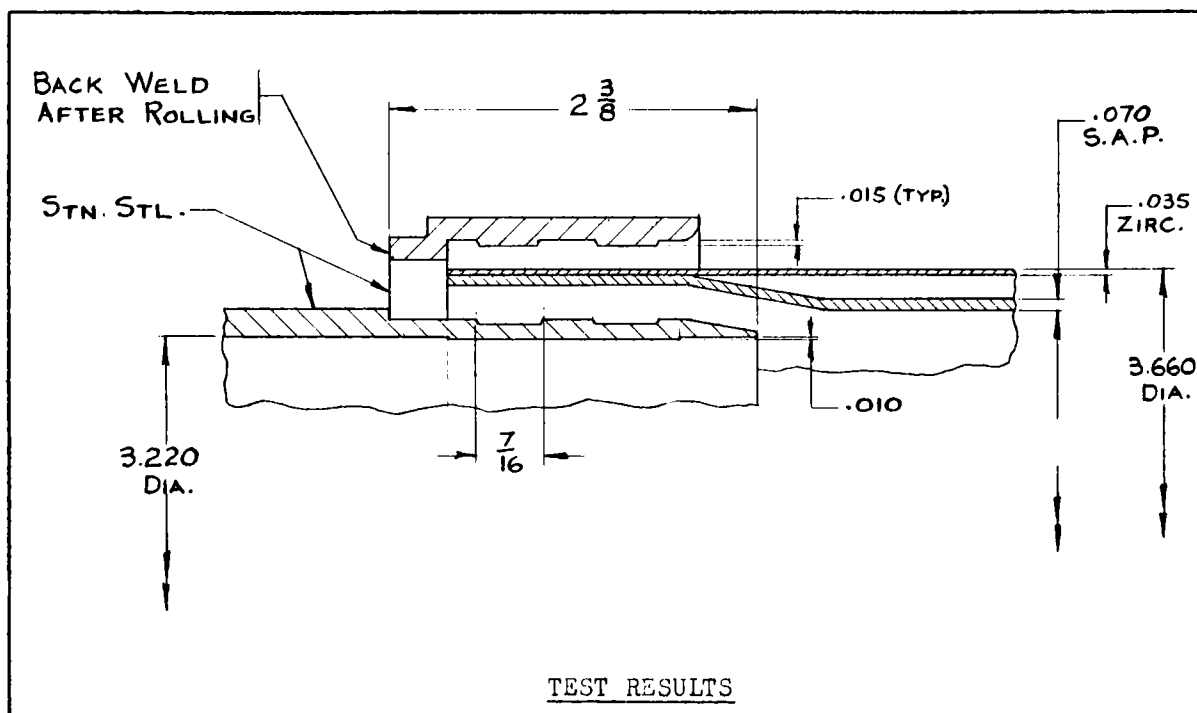
No.	Load Applied Between	Load 70° F	Load 700° F	Remarks
1	Stainless Steel - Zr II	8,000 lb.		Joint remained intact (estimated hydraulic load)
2	Stainless Steel - SAP	8,000 lb.		Joint remained intact (estimated hydraulic load)
3	Zr II - SAP	15,000 lb.		Joint remained intact (estimated hydraulic load)
4	Stainless Steel - Zr II	28,800 lb.	19,500 lb.	Failure occurred in the Zr tube remote from the joint
5	Stainless Steel - SAP	30,000 lb.	9,700 lb.	Failure occurred in the SAP tube at the edge of the joint

"CLUB SANDWICH" ROLLED JOINT

Figure 5 - "Club Sandwich" Rolled Joint

M-1 Fuel Channel

Solid Insulated
Fuel Channel
With Fluted
Liner

Solid Insulated
Fuel Channel
With Annular
Fuel
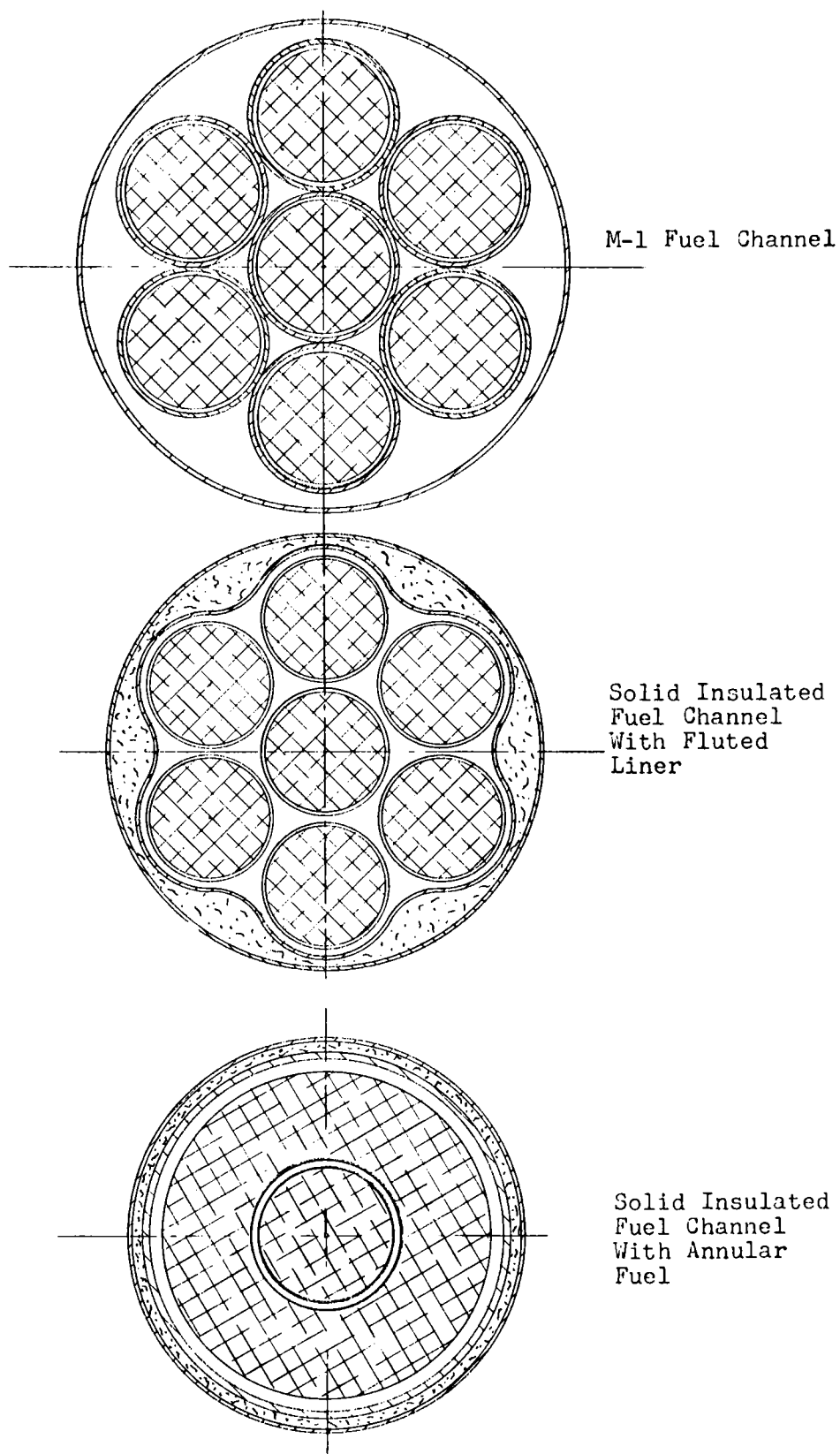
Figure 6 - Future Fuel Channel Cross Sections

REFERENCES

1. OCDRE Fuel Channel Technology by W. M. Brown
   Presented at the Euratom Joint Board Meeting - Ispra, Italy
   June 1961

2. OCDRE Solid Insulated Coolant Tube Assembly Stress Analysis
   CGE No. R61CAP17 by J. VanWinssen

3. OCDR Fuel Channel Development - Solid Insulated Assembly Liner
   Failures - J. D. Pritchard, R. W. Smith, E. J. Adams - CGE
   No. DF62CAP19

4. Development of Liner Tube Configuration for OCDRE Organic
   Insulated Coolant Tubes: Phase I Survey Study by R. L. Judd
   CGE No. R61CAP12

5. Development of Liner Tube Configurations for OCDRE Organic
   Insulated Coolant Tubes: Phase II Theoretical Analysis by
   J. T. Rogers - CGE No. R61CAP13

6. Development of Liner Tube Configurations for OCDRE Organic
   Insulated Coolant Tubes: Phase III Optimization Study by
   R. L. Judd - CGE No. R61CAP14

7. Liquid Insulation of Organic-Cooled Reactor Pressure Tubes by
   J. T. Rogers - Presented at the First Canadian Heat Transfer
   Symposium - Montreal, 15 June 1962

Fuel For Organic-Cooled Reactors

The Current Status at Chalk River - May, 1962

J. A. L. Robertson

Introduction

Most of AECL's work on developing fuel for the organic-cooled reactors is centered around the irradiation tests being performed at Chalk River. For the past two years, only the X-7 Loop of NRX has been available; the small diameter of its test-section has restricted the specimens to single elements and trefoils. The first testing of full-sized bundles, up to 10 cms (4 in.) diameter and 3m (10 ft) long, will be possible next year in the E-12 Loop of NRU. This loop is essentially a complete power-reactor fuel-channel in which the fuel, coolant and pressure tube can be irradiated together. It is designed for a power generation of 2MW, a maximum coolant temperature of 427°C and a maximum pressure of 42 kgf/cm² (600 psi). Another, small loop is being designed for a horizontal position in NRX. At least 50% of its volume will be within the reactor and a uranium annulus will surround the test-section to raise the fast neutron flux. Its main use will be to irradiate coolant to equilibrium content of high boilers, and not for fuel testing.

The fuel programme can be considered under three headings:-

1. UO_2 in SAP - the reference fuel for the OTR.
2. Alternative fuel - UC.
3. Alternative sheaths - beryllium, zirconium and graphite.

An outline of the reference design for the OTR fuel bundle is given in Appendix A. The irradiations already performed are itemised in Appendix B, and Appendix C provides some indications of the effort allocated. The last shows that the fuel work at Chalk River is largely confined to the actual irradiation and post-irradiation examination. Much of the supporting investigations and the preparation of specimens has been done by CGE at Peterborough. In addition, complete charges have been obtained from AI, the French CEA the UKAEA and Euratom.

UO_2 in SAP

The tentative specifications for OTR fuel (App. A) were arrived at from experience with water-cooled, Zircaloy-clad UO_2 and the irradiation tests in the X-7 loop of NRX have been performed mainly to prove the design. At the same time, they are intended to give the maximum design data for OCDR.

The tests to date (App. B) indicate that the design is acceptable, but emphasize the importance of rigorous quality

control of the SAP sheath and end closures. After an irradiation lasting 3-1/2 months there was no evidence of fuel/sheath reaction, and the interelement spacing of 1.0 mm had not led to preferential fouling at the gap. With the range of fuel/sheath diametral clearances tested (0.1 to 0.2 mm) and for values of $\int kd\theta$ up to 38 W/cm, no diametral expansion of the sheath occurred; at an $\int kd\theta$ of 46 W/cm, diametral expansions of 1/2% have been observed. Similarly, with dished pellets and axial clearances of 0.15 to 1% of the length, no sheath elongation greater than 0.1% has been observed even at the higher heat ratings. Although it proved possible to irradiate for 18 days a specimen with a deliberate hole in its sheath, the element was badly split when removed; decomposition of the coolant within the fuel results in a failure mechanism not experienced in water-cooled fuel elements. The evidence suggests that a defective element should not be subjected to power-cycling.

The next series of irradiations will test 122 cm (4 ft) long trefoils and in late 1962 full length, 244 cm (8 ft), trefoils will be tested. In addition to the reference sheath, a trefoil of elements each with 36 fins (fin touching fin) is to be included to determine the efficacy of the fins, which are calculated to reduce the surface temperature by about 80°C. Also under consideration is a series of experiments to define more closely the optimum diametral clearance. The E-12 (organic) loop of NRU is due to operate early in 1963, when a full-size, 19-element prototype will be tested at design ratings.

The calculations of mixing and subchannel temperatures have been partially confirmed using a salt-injection flow test at Chalk River. Full-size flow tests in organic coolant are to be conducted in an ex-reactor loop at Peterborough by CGE. Of particular interest will be the mechanical stability, vibration and wear of the complete OTR bundle.

Alternative Fuels - UC

The nuclear and physical properties of UC render it attractive as a fuel for organic-cooled reactors, but the fabrication costs commonly quoted would preclude its use in our application. G. H. Chalder (AECL) and C. A. Seger (CGE) have therefore studied the possible routes and analysed their potential costs. They concluded that the most attractive process was melting and casting of a pre-reacted (UO_2 + C) charge, and that fabricated, natural UC should be cheaper to produce than the same weight of comparable UO_2.

Accordingly, a development programme to investigate this route has been initiated at Eldorado Mining and Refining Limited. The initial work has shown that reacted sinters of adequate quality for subsequent melting can be prepared from pressed pellets of standard grade UO_2 and carbon (graphite or Sterling MT). The reaction was performed in an evacuated induction furnace.

Two loop irradiations have tested SAP-sheathed and steel-sheathed UC elements fabricated by Euratom; the steel-sheathed ones incorporated central thermocouples. Some failures of the SAP-sheathed elements during operation were probably attributable to weaknesses in the end closures, but incidentally provided an eight-day defect test for such specimens. Interpretation of the thermocouple readings

must await examination of the specimens, but tentative comparisons show that prior pressurization of the sheath onto the fuel significantly reduces temperatures in the latter. Proposals to irradiate 3 cm. diameter UC at 1kW/cm for one reactor cycle, and 1.4 cm diameter UC at 1.2 kW/cm to 5000 MWd/tonne U are being discussed with Euratom.

Alternative Sheaths - Beryllium, Zirconium alloys and Graphite

Eight specimens of beryllium-sheathed UO_2 were irradiated for one reactor cycle at power outputs up to 485 W/cm without incident. Diametral expansions of 0.5% occurred in specimens having 0.64 mm sheath and 0.03 mm fuel/sheath diametral clearance on assembly without causing failure. During the irradiation the fuel under went nine full-power cycles, but the coolant remained at elevated temperatures throughout. No further irradiations of beryllium sheathing are contemplated.

The possible use of zirconium alloys in organic coolant depends almost entirely on achieving acceptable rates of hydriding. Following corrosion tests on Zircaloy samples, two Zircaloy-sheathed UO_2 fuel elements were irradiated for three weeks at 430°C. Subsequent analyses have shown a hydrogen pick-up of <10 ppm, but a surface layer, tentatively identified as ZrO_2, 0.03 mm (0.0015 in) thick has been observed metallographically. The projected use of a Zircaloy liner tube for the in-reactor section of the loop should yield further relevant information.

Graphite is being considered as a potential sheathing material for uranium carbide. So far, it has been shown that impermeable grades can be brazed using a variety of alloys and the resulting seals are to be leak-tested. The possibility of coating and sealing in one operation sheaths of normal reactor grade graphite is to be investigated. Laboratory corrosion tests of brazed graphite are under way and loop irradiations of experimental fuel elements are intended for 1963.

APPENDIX A

REFERENCE DESIGN FOR THE OTR FUEL

Fuel: Stoichiometric, sintered UO_2 pellets.

Sheath: SAP of wall thickness 0.63 mm (0.025 in.), with

three fins machined from thick wall tube.

Element dimensions: 1.53 cm (0.6035 in.) O.D. by 2.42 m

(8 ft) long, i.e., full core length.

Inter-element spacing: 1.0 mm (0.040 in.)

Max. fuel power / length*: 460 W/cm (48,000 BTU/ft .h)

Max. $\int k.d\Theta$*: 36W/cm (3,800 BTU/ft.h)

Max. surface heat flux* : 95 W/cm2 (300,000 BTU/ft2.h)

Max. local sheath temperature* : 480°C (895°F)

Max. burn-up: Estimated to be about 5000 MWd/tonne U.

* Maximum values for normal, full-power operation; during
over-power transients values will be higher.

APPENDIX B

IRRADIATIONS IN THE X-7 ORGANIC LOOP OF NRX

Test No.	Brief Description	Duration	Reports Available
1. UO$_2$ in SAP			
X-57	Loop-commissioning & Zircaloy-feasibility test	27July-19Sept 60	Draft by Cracknell MacDonald, Lew and Heal.
X-58	1st test of SAP-clad UO$_2$	18Oct-21 Nov 60	CRFD-1084 by MacDonald & Aspila
X-59	1st defect test of SAP-clad UO$_2$	22Nov-14 Dec.60	AECL-1390 by Robertson, Cracknel & MacDonald
X-708	Al small finned elements	2Mar-Apr.61	Draft by Lew & MacDonald
X-704 Phase 1	1st test of SAP/UO$_2$ trefoil	5Apr-7Apr 61	
Phase 2	Repeat	25Apr-29May 61	
X-709 Phase 1	Co-operative, long-term trefoil test	14Aug-22Nov.61	ANS paper by Robertson, Thexton & Lew
Phase 2	Repeat	23Mar-24Mar 62	
2. Alternative Fuel - UC			
X-711	UC charge: 2 central T/Cs	22Feb-6Mar 62	Exp-NRX-71102 by MacDonald & Draft by May & Stoute.
X-715	Effect of pressurised sheath on UC temp.	25 Apr-22May 62	
3. Alternative Sheaths - Be, Zr and Graphite			
X-67	Zircaloy corrosion test	20Sept-18Oct 60	Exp-NRX-6705 by Krenz.
X-707	Be-and Zircaloy-clad UO$_2$	29Jan-19Feb 62	Draft by Greetham

APPENDIX C

SUMMARY OF EFFORT ALLOCATED

1. UO₂ in SAP

AECL development contract with CGE, Peterborough -1962 estimates

Fuel design	$12,706.00
Manufacturing, quality control and Metallurgical engineering	6,902.00
Thermal , strength, hydrodynamic, vibration & wear tests	54,061.00
Fabrication development	102,645.00
Fabrication of fuel for irradiation	52,206.00
Assisting irradiation & examination at Chalk River	16,820.00
OTR fuel	27,060.00
Tech. supervision, reports etc.	36,217.00
	308,617.00

2. Alternative Fuel - UC

AECL development contract with Eldorado M & R Ltd. - 1962 estimate

UC fabrication, Capital equipment (EMR Budget)	60,000.00
Operation	60,000.00
	120,000.00

Not included in the above contracts are the following services provided by AECL at Chalk River:-

Operation of reactor and loop, fabrication of enriched UO₂ pellets, direction and design of irradiation, post-irradiation examination, metallography and chemical analysis.

AECL effort on alternative fuel and sheaths for organic-cooled reactors is integrated with similar work for other systems, but about five men (scientists or technicians) would be employed full-time.

X-706 COOLANT CONDITIONS

JUNE	PCFT mg	FPFT mg	Fe ppm	Water ppm	Gas Content cm^3/kg					
					H$_2$	N$_2$	CH$_4$	CO	Hydro- carbons	Total
2	74									
3				182						
4	91									
5	60		8							
6					13	168	8	1.8	31	224
7		69								
8	83	71	3							
9										
10										
11										
12	102		3	163						
13					22	206	19	0.9	41	289
14		50								
15	103	143								
16										
17										
18										
19	124		3							
20		71			59	107	10	1.6	-	-
21		52								
22	78	51	5							
23		87								
24										
25										
26										
27	104									

Fuel Element	BJO	BJP	BJR	BJS	BJT	BJU	BJV
Velocity, m/sec	7.1	7.1	7.1	2.5	2.5	2.5	2.5
Reynolds no.	52,000	52,000	52,000	43,000	43,000	43,000	43,000
Pressure at outlet kg/cm^2	17.1	-	-	-	-	-	18.6
Bulk temp.,°C	356	353	350	348	347	347	346
Calculated surface temperature,°C	420	427	441	416	415	397	374
Av. heat flux , w/cm^2	65	76	93	25	25	19	10
Est'd beta current, μamp/cm^2	0.01	0.02	0.2	0.06	0.007	0.005	0.03
Duration, hr.				496			

FINAL DEPOSIT

Thickness, μm

	BJO	BJP	BJR	BJS	BJT	BJU	BJV
average	46	51	30	25	20	20	20
min.	0	38	-	20	0	0	20
max.	51	51	-	38	25	30	20
Weight, mg/cm^2	15	16	21	13	7	5	5
μg/cm^2/hr	32	34	44	27	15	11	11
%Loss on ignition	9	-	11	11	6	11	34
% Fe	58	-	58	62	68	58	37
Fe content of film, mg	600	640*	840	560	330	200	130

Total 3300 mg
≤10 ppm Fe in coolant

Fe compounds			oxide + trace <Fe		oxide + trace Fe		oxide + trace <Fe
Elec. resistance	Low	-	Low	-	-	Low	-

* Assumed 58% Fe in deposit

SUMMARY OF OUT-REACTOR FOULING TESTS

Coolant Conditions

	1	2	3	4	5	6	7	8 (No.7+B-SA)	8 (No.8-1)	8 (No.8-3)	9
Source	B*	No. 1 + B	X-7	No.3	No. 4 + X-7	B - SA	No. 6 + B - SA	No.7+B-SA	No.8 - 1	No.8-3	No. 8 - 2 + B-SA
Pretreatment	F*	F	None	None	10g. 8-quinolinol added	F	D - RA - F	D - RA - F	W	W	D - RA - F
Treatment during Test	F*	F	None	None	None	F	RA - F	RA - F	RA - F	None	RA - F
Ash, ppm	-*				-	2	1	3	4	6	3
Fe, "	16	3	2	2	-	<2	<2	2		3	
Water, "	176*	214	250	220	-	87	40	59	210	240	30
PCFT, mg	160*	65	150	70	-	13		37	9	10	9

Operating Conditions and Deposit

	1/2A	1/2B	1/3A	1/3B	2/2A	2/2B	2/3A	2/3B	3/2A	3/2B	3/3A	3/3B	4/2A	4/2B	4/3A	4/3B	5/2A	5/2B	5/3A	5/3B	6/2A	6/2B	6/3A	6/3B	7/2A	7/2B	7/3A	7/3B	8/1A	8/1B	8/3A	8/3B	8/2A	8/2B	9/2A	9/2B	9/3A	9/3B
Test duration, hr	625	625	625	625	556	556	556	556	478	478	478	478	478	478	478	478	71	71	71	71	281	281	281	281	348	348	348	348	163	163	190	190	140	140	44	44	44	44
Velocity, m/sec	1.5*	1.5*	3.7*	3.7*	3.8	3.8	1.7	1.7	1.3	1.3	3.0	3.0	1.3	1.3	3.0	3.0	1.3	1.3	1.3	3.1	3.1	3.1	7.2	7.2	7.2	3.1	3.1	3.1	3.1	7.2	7.2	7.2	7.2	7.2	7.2	7.2	7.2	7.2
Initial surf.temp.°C	504	536	489*	527*	438	480	446	485	397	416	414	431	415	436	427	446	452	484	453	482	457	484	459	482	487	491	485	490	484	489	477	476	476	481	416	438	476	496
Coolant temp.,°C	344	351	357	364	267	279	288	298	344	348	353	357	342	347	352	357	343	349	356	364	345	352	358	365	347	354	360	367	347	353	348	348	348	356	345	351	357	365
DEPOSIT Texture	S	StN	S	StN	L	L	S	S	SX	SX	CX	CX	S	S	CX	CX	S	C	S	SX	S	S	CX	CX	S	S	S	S	S	S	SX	SX	SX	SX	StCX	CX	CX	CX
Max. thickness, µm	<3	13	<3	8	2	7	3	11	-	30	-	50	-	40	-	70	30	60	-	50	4	10	25	25	-	-	-	-	-	-	-	-	-	-	*	*	*	*
Deposition rate, µg/cm²-hr	-	4	-	-	0.9	2.5	2.2	4.3	3.6	6.1	12	16	6.3	13	13	24	44	111	63	142	3.2	6.8	7.8	9.6	0.9	0.9	0.6	0.9	0.9	1.0	1.7	1.6	3.8	3.8	18	47	109	144
Change in thermal resistance (watts/cm²-°C)⁻¹	-	-	-	-	0.07	0.13	0.0	0.12	1.6	2.7	-	-	1.6	2.2	0.4	0.6	-	1.4	-	0.9	0.09	0.30	-0.37	-0.38	0.03	0.03	0.01	0.01	0.0	0.0	-0.04	-0.04	-0.07	-0.08	-0.34	-0.47	-0.46	-0.53
Residue on ignition, wt.%	-	14	-	-	-	5	-	9	101	92	101	100	88	73	104	99	100	67	78	57	-	100	109	100	-	-	-	-	-	-	-	-	-	-	-	106	111	119
Fe compounds	-	0	-	-	-	0	-	0	0,I	0,I	0,I	0,I	0,I	0,I	0,I	0,I	100	-	I	-	-	I,0	-	0	-	-	-	I	-	-	0,I	-	0	-	0,I	0,I	0,I	0,I
Elec. resistance (arbitrary units)	$>10^6$	$>10^6$	$>10^6$	$>10^6$	$>10^6$	$>10^6$	$>10^6$	$>10^6$	1	3	1	1	1	$>10^6$	1	2	100*	$>10^6$	1000*	$>10^6$	0.5	40	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5

	1	2	3	4	5	6	7	8	9
Total Fe in deposit / Total Fe in coolant	~0.1	~0.3	11	13	-	6	3	0.1 / 0.2 / 0.3	10

SYMBOLS:

B = Blended o-, m-terphenyl + 30% OMRE HB, rough filtered through 10 µm filter
SA = Single pass through Attapulgus clay bed
F = Recycled through 5µm sintered stainless-steel filter
RA = Recycled through Attapulgus clay bed
D = Dried with Molecular Sieve bed
W = Water added
X-7 = Removed from X-7 loop
* = Variable, see text for details.

Deposit:
S = Smooth, fine-grained
C = Coarse-grained
L = Very smooth, lacquer-like
X = Perceptible crystals
N = Nodules
O = Iron Oxide
I = α-iron

-65-

SUMMARY* OF X-7 IRRADIATIONS

Test	Fuel Elements				Av.* Irradiation Conditions				Av.* Coolant Conditions			Deposit*		
	Arrangement	No. of Elements	Sheath	Fuel	Duration days	Velocity m/s	Bulk Temp., °C	Surface Temp., °C	Iron ppm	Water ppm	PCFT mg	Wt/unit area mg/cm²	Thickness µm	% Residue on ignition
X-57 Commissioning	Single string	7	S.S.	UO$_2$	44	9	385	460	3	-	-	-	25 – 100	80
X-67 Zircaloy Feasibility	Single string	2	S.S.	UO$_2$	20	9	260	370	3	-	-	-	<10	-
X-58 First SAP	Single string	7	SAP	UO$_2$	0.3	9	385	475	18	-	-	-	<5	-
X-59 SAP Defect	Single	1	SAP	UO$_2$	15	9	370	475	3	-	-	-	<10	-
X-708 Finned Small Element	Single string	7	SAP	UO$_2$	30	9-4.	370	435	3	230	75	-	20 – 35	-
X-704-1 First Trefoil, Phase 1	2 Trefoils / 1 Trefoil	6 / 3	SAP / S.S.	UO$_2$ }	1	8	330	420	7	190	-	-	10 – 500	-
X-704-2, First Trefoil, Phase 2	1 Trefoil / 1 Trefoil	3 / 3	SAP / S.S.	UO$_2$ }	32	8	340	430	4	150	70	-	10 – 1400	53
X-706 Fuel Element Fouling	Single String	4 / 3	S.S. / S.S.	Pu-Al / Pu-Al	21 / 21	2.5 / 7	347 / 353	400 / 430	3 / 3	170 / 170	90 / 90	7 / 17	0 – 40 / 0 – 50	90 / 90
X-709-1 Long Term Trefoil, Phase 1	2 Trefoils	6	SAP	UO$_2$	76	7	310	450	3	170	10	-	40 – 80	70
X-707 Beryllium-Zircaloy Clad	Single string	10	8-Be; 2 Zr.	UO$_2$	20	9	375	450	4	50	6	0.2	<5	-
X-711 Uranium Carbide	Single string	8	6SAP; 2 S.S.	UC	8	8	340	410	2	30.	4	-	<5	-
X-709-2 Long-term Trefoil, Phase 2.	2 Trefoils	6	SAP	UO$_2$	0.8	7	295	420	2	35	4	0.02	<5	-

* Note: The values given in this table are, in many cases, representative aver
-ges & should not be used as a basis for quantitative conclusions.

FIGURE 1.
OUT-REACTOR FOULING TEST No. 9
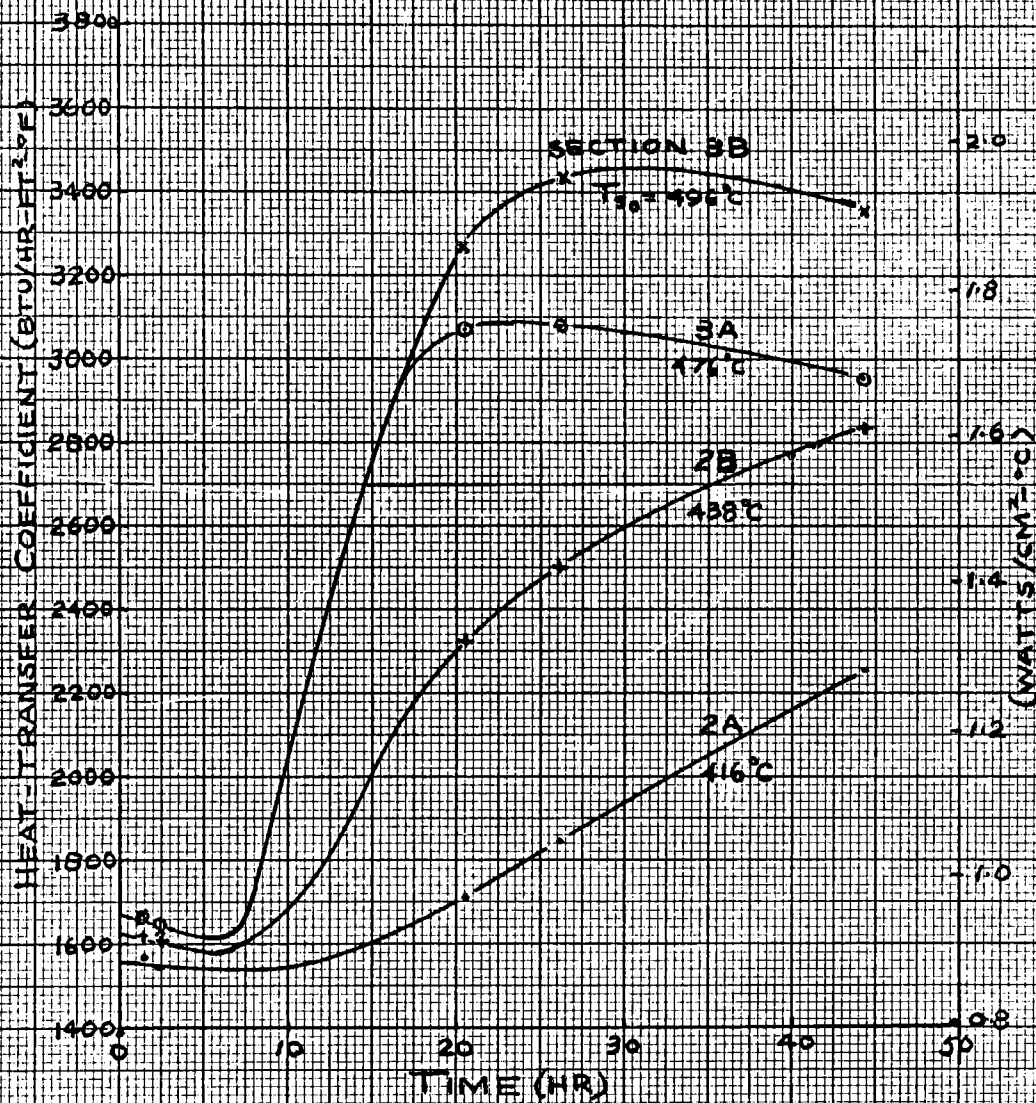
THE DEVELOPMENT OF MATERIALS FOR AN OCDR

by

W. R. Thomas

Atomic Energy of Canada Limited
Chalk River, Ontario

Two types of material are under intensive investigation for application in the core of an organic-cooled power reactor; SAP and zirconium alloys. Investigations are being carried out at Canadian General Electric, Canadian Westinghouse, Orenda Engine Division of Hawker-Siddeley and at AECL. The work going on for each type of material will be summarized for these establishments.

SAP

Canadian General Electric

Most of the work on the procurement and evaluation of SAP is being carried out at CGE. Their objectives are

(1) to encourage suitable sources of high quality raw material
(2) to develop suppliers of tubing of high quality and integrity
(3) to develop methods of joining SAP to SAP for the fabrication of end closures for fuel elements.

The only important sources of raw material are AIAG, Switzerland and the Aluminum Company of America. Materials from both of these companies have been evaluated as well as a small amount of material from Trefileries et Laminoires du Havre.

The results of most of this work up to the beginning of this year has been reported in "Mechanical Properties of Dispersion Strengthened Aluminum Alloys" R61CAP69 by Boxall and Standish, and in an earlier report "The Present Status of SAP in the Canadian Organic Cooled Reactor Program" R61CAP30, June 1961 by Boxall and Mocarski.

The results of the tests showed that marked directionality of properties was present in the billets and that non-uniform distribution of oxide might be a contributing factor.

Also given in these reports are the results of tubing evaluation. Tubing suppliers have been ALCOA, General Impact Extrusions, Imperial Chemical Industries and High Duty Alloys. Both finned and plain tubing have been stress-rupture tested by internal pressurization. The results showed that the 100 hours stress to rupture was somewhat better for tubing than for the bar stock from which it was made. This was attributed to the difference in stress systems.

The only method of joining which has been examined at CGE has been hot pressure bonding. The main purpose in pursuing this method was to provide a means for the fabrication of loop specimens. It is intended to investigate magnetic force welding as a production method. A suitable machine is being purchased and will be in operation before the end of the year.

The amounts being spent on this program are summarized in the following table:-

Project	Spent	Budget (1962)
SAP material inventory Evaluation, testing and analysis	20,000.00	94,467.00
Procurement and tubing fabrication development	1,600.00	111,000.00
		$ 205,467.00

Of this sum $10,000 is for material, therefore, the remainder represents 8 professional man-years.

AECL

The effect of neutron irradiation on the mechanical properties of SAP has already been studied at Chalk River and reported in CRMet-983.

Further irradiations are planned to larger neutron doses in the NRU reactor. When the new fast neutron facility becomes available in NRU it will be possible to study the effect of neutron irradiation on stress-rupture properties. This will probably be toward the end of 1963.

ZIRCONIUM ALLOYS

Canadian Westinghouse

Canadian Westinghouse are looking into the possibility of using zirconium alloys for pressure tube material in organic-cooled reactors. This necessitates the use of hydrogen barriers on the surface of the pressure tubes. Thin layers of aluminum and zirconium oxide are at present being given first consideration, although Ni and Nb coatings will also be tested.

Hydrogen pickup rates on specimens having various surface layers are being carried out in an organic autoclave. The initial test conditions are as follows:

Coolant temperature	- 400°C
Hydrogen pressure	- 1 atmosphere
Water content of organic	- as received ($\sim$ 100 ppm)
Duration of each test	- 2 weeks.

The specimens with the barrier layer showing the lowest hydrogen pickup will then be tested at two additional hydrogen pressures in order to permit extrapolation to reactor hydrogen pressures. If a decision is made to use zirconium oxide as a barrier the effect of water content of the organic on hydrogen pickup will also be investigated.

A total of $57,000 has been allocated to this program.

Orenda Engines

Although aluminum looks promising as a hydrogen barrier from the point of view of hydrogen diffusion, its direct application to zirconium alloys seems to be undesirable. Interdiffusion between Zr and Al at 400°C is quite rapid, resulting in the formation of brittle intermetallics. An intermediate barrier between the Al and Zr may therefore be necessary.

Orenda Engines are looking into the possibility of using either Ni or Nb as an intermediate barrier. Although Ni does not look favorable because of fairly rapid interdiffusion with Al, it is being used in the preparation of the initial specimens for testing at Westinghouse.

Nb looks like a good material for the intermediate barrier both from point of view of neutron economy and compatibility with Zr and Al. Applying it may, however, be difficult. It is proposed to deposit Nb onto Zr by the reduction of Nb halide. No difficulty is anticipated in depositing the Al onto Nb, with probably electroplating or electroless plating being the method used. A total of $20,000 has been allocated to this program.

A Zr-Al-Mo alloy will be assessed to determine its mechanical properties and creep strength at temperatures up to 425°C. Published results indicate that the properties will allow a design stress of 15,000 psi at this temperature. The results of the barrier layer tests on Zircaloy should be equally applicable to this alloy.

AECL

Tests in the organic loop (X-7) have indicated that the hydrogen pick-up from the coolant by Zircaloy may not be as serious a problem as was first supposed. Samples exposed for two cycles at 260°C showed no increase in hydrogen concentration. Zircaloy-clad fuel elements which had been autoclaved have been irradiated in the X-7 loop for 3 weeks at a surface temperature of 460°C with no measurable increase in hydrogen concentration. It is planned to install an autoclaved Zircaloy test section in the NRU E-12 loop and in the X-7.

This part of the program occupies 1 professional full time.

ZIRCONIUM ALLOYS IN ORGANICS: A POSSIBILITY

by

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I. INTRODUCTION

Zirconium alloys are foremost among materials of construction for reactor cores, yet they have not been considered suitable for use in reactors employing molten organic coolants because of the unfavourable results of early corrosion experiments. A survey of the published data shows, however, that there may be scope for development.

I. 1 Results of Thermal Tests:

Barker and Mausteller[1] have reported that Zircaloy-2 and Zircaloy-3 in molten biphenyl at 427°C for 30 days became severely hydrided. In runs at coolant velocities greater than zero ft/sec the samples had flaked and broken. Work at AERE[2] showed that zirconium was badly attacked in "Santowax R"* at 400°C. In Santowax at 300°C, corrosion was very slight; no significant weight changes were found after 30 days. The effects of variation in temperature and H_2 partial pressure were not studied, however.

Troutner[3] noted that at 400°C and under a hydrogen partial pressure of 0.1 atm, zirconium was severely hydrided in MIPB.** Between 700 and 8700 ppm H was picked up in 6 days. At lower temperatures this effect was much less severe, and at 250°C the pick-up in six days was insignificant. The effect of H_2 partial pressure was also apparent since, even at higher temperature, a five-fold reduction in H_2 partial pressure caused a three-fold to twenty-fold reduction in H pick-up. These results refer to MIPB containing 60 ppm H_2O. Troutner also found that increasing the H_2O content to 100 ppm and 150 ppm caused significant reductions in the H pick-up. His results are summarized in Table I.

The above results resemble those reported by Boyle and Kisiel[4] who found that 1 part of water vapour in 10^4 parts of hydrogen gas inhibited H pick-up by Zircaloy-2 exposed in the gas at 400°C and 1 atm pressure. This effect is undoubtedly related to the protection afforded by the film of oxide formed on zirconium alloys by the water

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* "Santowax R" = 16% ortho - 53% meta - and 30% para-terphenyl
+ biphenyl + tar ($\sim$ 1%)
** MIPB = monoisopropylbiphenyl

Troutner[3] did not report weight changes of his specimens. The relation of hydrogen pick-up to weight change is shown in Table II, in which AERE and BMI data[5] are set forth. With notably few exceptions extensive hydriding corresponds to high weight gain; but it should be pointed out that the actual weight of hydrogen picked up does not account for more than a small fraction of the weight change. *

The data confirm the dependence of hydrogen pick-up on temperature and hydrogen overpressure already noted, but the trend with increasing water content (Table I) is apparently reversed. Unfortunately, the technique of the water determinations has not been developed to the point at which results from different laboratories can be usefully compared. However, the results at $400^{\circ}C$ -$427^{\circ}C$ in Table I and II appear to indicate minimum hydrogen pick-up at a water concentration of 100-150 ppm, if it is assumed that the different coolants are comparable in their behaviour.

I. 2 Experiments with Samples under Irradiation:

The effects of radiation on an organic coolant may be likened to the effect of momentarily raising microscopic volumes of the coolant to very high temperatures. Thus, even though the bulk temperature of the coolant is low, we may expect irradiation to produce effects resembling those produced thermally at much higher temperatures.

Available US data on corrosion of zirconium in organic coolants under irradiation are limited to early experiments of Kline, Gioseffi and Bley[6]. Their results are summarized in Table III. Unfortunately, no information regarding hydrogen overpressure or water content is available. It will be seen, however, that the weight gain and hydrogen pick-up of these specimens are generally in accord with results in Table II. The hydrogen does not account for more than a small fraction of the weight changes. The hydrogen pick-up is averaged for three specimens in each of the two positions, and the difference between unirradiated and irradiated specimens would appear to be significant. Hydrogen pick-up does not, in this experiment, appear to be sensitive to radiation. It is probable that the radiation intensity was too low to have an appreciable effect.

Table IV summarizes some early Chalk River results with Zircaloy-2 in the first NRX Organic Coolant Loop[7]. The in-reactor test section of this loop was installed in a small instrumental hole, I-2-A, tangential to the NRX calandria. Though no H-determination was made

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* For example, if a weight gain of 45 mg/dm^2 had been produced by H pick-up in Davies' samples, the hydrogen content of the samples would have been $\sim$3000 ppm. Only about 200 ppm was found, however.

on samples which had been given the autoclave treatment only, it is probable that 18-20 ppm of hydrogen was introduced by this treatment. The results therefore indicate negligible H pick-up under the conditions of this experiment, and no effect of radiation can be detected. Samples were weighed at several intervals during the test, but no consistent weight changes could be detected.

I. 3 Summary

It is by no means clear that zirconium and Zircaloy-2 are incompatible with organic coolants under all conditions. The above review shows that the coolant composition, the temperature and possibly the alloy composition play a part in determining the attack on the alloy by the coolant. At least three, and possibly four, approaches to the use of zirconium alloys in these media can be suggested: (a) insulate the alloy so that it does not come into contact with coolant at temperatures above about 250°C, (b) coat the alloy with a material which is relatively impervious to hydrogen, (c) control coolant conditions so that a protective oxide, which provides a barrier to the entrance of hydrogen, is maintained on the surface of the alloy. Another, though less likely possibility is to reduce the hydrogen pressure to a value at which the rate of hydrogen pick-up by the alloy becomes tolerable.

This report describes the results of experiments at Chalk River based on the first three of these approaches. Some tests have been performed to determine the feasibility of using a zirconium alloy in an insulated, stressed coolant flow tube; others have been made on various metallic barriers to hydrogen, as applied to zirconium alloys. The report is a progress report, since much of this work is going on at the time of writing. The results are presented however, since they show that zirconium alloys are by no means ruled out for use with organic coolants, and that there is considerable scope for development.

II. APPLICATION OF ZIRCONIUM ALLOYS TO INSULATED COOLANT TUBES IN AN ORGANIC COOLED REACTOR

Assuming that difficulties in fabrication and installation are satisfactorily overcome, the chief problem in the use of a thermally insulated zirconium alloy coolant tube is that which arises if, for any reason, the insulation fails. The zirconium alloy then comes into contact with hot organic coolant in an intense radiation field. The problem has been considered by H. K. Rae and D. W. Charlesworth[8] for the insulated coolant tube shown in Fig. 1. Here a zirconium alloy stress tube is in contact with the moderator at 82°C. This is a hypothetical design and bears no relation to any actual Canadian organic reactor.

Three types of insulation failure can be considered:

(1) In most cases there will be a small local failure of the insulation and a seepage of coolant to the inner coolant tube surface. The temperature of this surface may then rise to between 100 and 150°C.

(2) A large crack may appear in the insulation and allow coolant to flow in intimate contact with the inner coolant tube surface. The temperature may then rise as high as 260°C, at which value it is stabilized by nucleate boiling of the moderator. This is considered by Rae to be the reasonable upper limit of temperature which could possibly be experienced by the coolant tube.

(3) If, however, a large section of the insulation should be lost, so that two or more square feet of the inner surface of the coolant tube should be exposed to the coolant, then film boiling of the moderator might occur and a new temperature profile would be established. The maximum temperature experienced by the coolant tube would then be ~370°C.

The temperature profiles representative of these three conditions are shown in Fig. 2. The experiment described below was intended to show whether Zircaloy-2 would survive an exposure to molten organic coolant at 260°C for a reactor cycle of 4 weeks without picking up excessive amounts of hydrogen. Another alloy, Zr-0.5 Cu-0.5 Mo, was also tested. The conditions were intended to simulate the situation in which the insulation of an internally insulated pressure tube breaks down sufficiently to admit hot organic liquid to the zirconium alloy surface, this being the worst type of failure in which the outcome would be affected by hydriding of the Zircaloy-2.

II. 1 <u>Materials</u>

The Zircaloy-2 used in this test was fully annealed, 1.4 mm thick, reactor-grade strip from Mallory-Sharon Metals Corp.* ingot number 28094.

The Zir-0.5 Cu-0.5 Mo alloy was received from A. I. John Thompson, Ltd. as hot-rolled slab. This was cut into bars 5 mm thick and cold rolled to 1.4 mm strip. The available analytical data are summarized in Table V. The Zr-0.5 Cu-0.5 Mo alloy is referred to hereafter as the "ATR" alloy.

The ATR strip was pickled to remove possible surface contamination, then annealed in vacuum at 750°C for 30 minutes. Small, flat

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* Now Reactive Metals, Inc

tensile specimens in both alloys were punched from the corresponding strips. These were trimmed by filing, marked with identification numbers and flash pickled to remove any iron picked up from the tools. They were then vacuum annealed at 450°C for 1 hour, given a final flash pickle, carefully rinsed, dried and measured.

A few specimens in each of the two alloys were then autoclaved in steam at 400°C and 100 atm for 148 hours to coat them with an oxide film. A random selection of specimens representative of the autoclave and the freshly pickled alloys were assembled in the specimen holder.

II. 2 Specimen Holder

A sketch of the specimen holder and assembled specimens is shown in Fig. 3. To facilitate design and handling, the specimen holder was designed as a self-contained unit to fit inside the flow tube in the reactor loop test section. The assembly was designed to provide a coolant velocity of 9 m/sec over the end portions of the test specimens. It was designed for ease of disassembly and inspection in a hot cell.

II. 3 Loop Exposure:

The samples were irradiated in X-7 loop for 24 days in Santowax OM plus 30% OMRE high boilers. The specimen holder (Fig. 3) was attached to a fuel string at the position of maximum flux in the test section. The temperature of the Santowax during the test averaged about 252°C on the inlet side of the test section and 256°C on the outlet side. The reactor was at 40 MW for all but about 48 hours of the exposure. An estimate of the average energy released in the organic coolant in the test section at full reactor power is 150 mW/g ± 50 mW/g.

The water content of the Santowax during this period was determined by azeotropic distillation followed by Karl Fischer titration. The analyses are now in doubt, as subsequent studies of analytical techniques have shown that the results were high due to contamination with atmospheric humidity. Present estimates are that the water content could have been in the range from 100 ppm - 900 ppm, but most probably about 100 - 150 ppm.

Analyses made on the hydrogen content of the Santowax during the run showed that amounts of hydrogen varied between 6 cm^3/kg to 15 cm^3/kg - equivalent to partial pressures of 0.05 - 0.13 atm.

II. 4 Experimental Results:

The measurements of the weight changes of the specimens are summarized in Table VI; Table VII summarizes the results of the hydrogen pick-up measurements.

The values for the hydrogen contents of the as-pickled ATR alloy are at the limit of detection and the reported pick-up is probably not significant.

II. 5 Discussion:

The Zircaloy-2 samples used in this experiment behaved normally during the prefilming exposure in steam with the exception of Z-1 which appeared to have gained less weight than the others. A normal weight gain of 21 mg/dm^2 corresponds in these samples to a hydrogen pick-up of 25 - 30 ppm. The pick-up found was actually somewhat higher, being 35 ± 8 ppm. When these autoclaved specimens were put into the loop there was no significant weight gain (Table VII). However, there was a significant increase in hydrogen content of about 13 ppm. Had this resulted from reaction with water in the Santowax, there should have been an equivalent weight gain of 10 mg/dm^2. Therefore, the hydrogen must have been absorbed from the Santowax. On the other hand, the Zircaloy-2 samples which were put into the loop in the as-pickled condition, and which also showed no significant weight increase, picked up less hydrogen than the autoclaved samples.

An effect similar to this has been reported for zirconium exposed to H$_2$ gas at 150°C and 24 mm pressure. Gulbransen and Andrew[9] found that the oxide film formed at room temperature on a freshly pickled surface has a much lower permeability to H$_2$ than an oxide film formed at 275°C. Their test were conducted over a period of only two hours, but their hydrogen pressure of 24 mm was considerably higher than the equilibrium partial pressure of hydrogen of about 1 mm of Hg over a solution of about 15 cm^3/kg in Santowax.

The higher weight gain of the ATR alloys in the preliminary autoclaving was not abnormal for the test conditions used. The samples came out of the autoclave with a smooth black oxide film with brownish overtones. The weight loss shown by the oxidized specimens in the loop is unaccountable at the time of writing. The average hydrogen content of these specimens did not change, though there was a larger spread of values in the irradiated material. The samples which started with a pickled surface gained slightly in weight. The increase in the average hydrogen content is of doubtful significance in view of the large spread of results. The indications are that the ATR alloy is susceptible to greater weight changes, i. e , changes in the oxide film, but less hydrogen pick-up than the Zircaloy-2

The ATR alloy was originally chosen because of its known superior corrosion resistance in high temperature steam. Therefore, if the effects of residual water in the organic coolant were important, it was expected that the ATR alloy might give a better performance than

Zircaloy-2. After the experiment had been started, however, it was established that the oxidation resistance of the ATR alloy was inferior to that of Zircaloy-2 in steam or water below about 400°C and that its corrosion resistance improved only at higher temperatures. Thus, if the weight increase of the pickled ATR alloys is an indication of a reaction with residual water in the Santowax, it is consistent that this increase would be greater than for Zircaloy-2.

There appears to have been a much better barrier to hydrogen on the surface of this alloy than on the Zircaloy-2.

III. HYDROGEN BARRIERS ON ZIRCONIUM ALLOYS

A preliminary investigation has been made of the feasibility of using zirconium alloys in organic-cooled reactors provided the alloy is clad with a material highly impermeable to hydrogen. Among the common metals, aluminum is most impermeable to hydrogen[10]. It has been fairly well established[11, 12] that the passage of hydrogen through aluminum is controlled by diffusion through the surface oxide. This fact, combined with the good stability of aluminum-oxide suggest that very thin layers (≈ 0.005 cm) of aluminum will offer adequate protection against hydrogen pick-up. The low neutron cross-section (0.23 barns) also makes aluminum a desirable material for reactor applications. Aluminum was therefore given first consideration as a cladding material.

Type 304 stainless steel was also tested as a possible hydrogen barrier. Although preliminary calculations were not favourable, the ease of fabrication of stainless steel clad tubing (by co-extrusion) made a limited investigation desirable.

Some results have been obtained on the protection offered by pre-formed zirconium-oxide and oxide formed during exposure to the organic.

The permeabilities of zirconium and aluminum to hydrogen seem to decrease markedly for at least several hours as a result of being heated in helium[12]. To check this effect for longer times some specimens were given a pre-treatment in helium.

III. 1 Experimental Details:

The Zircaloy-2 used in the following tests was in the form of sheet 0.16 cm thick obtained from the Wah Chang Corp. Some of this was subsequently rolled down to 0.1 cm. All specimens were annealed in vacuum ($\sim 10^{-6}$ torr) at 750°C for one hour. This reduced the

hydrogen content of the material to about 3 ppm and also served as an annealing treatment. The available analytical data are summarized in Table V.

The aluminum and stainless steel claddings were not directly bonded to the Zircaloy-2 specimens. Because of ease in fabrication the specimens were made in the form of sandwiches as shown in Fig. 4. The peripheral welds were checked for leaks by the usual liquid air-alcohol test.

Unclad Zircaloy was given one of the following surface treatments prior to testing in organic:

a) abraded
b) autoclaved for 3 days in 6.9 atm steam at 400°C
c) heated in one atmosphere of helium for 6 hours at 600°C
d) heated in air at 800°C for 1/2 hour
e) autoclaved + helium treatment

Some of the aluminum-clad sandwiches were heated in air for 4 hours at 600°C. Of these, several were further heated in helium for 6 hours at 600°C.

Two tests were run in a small stainless steel bomb. The Zircaloy discs were 1.25 cm in diameter by 0.158 cm thick. The sheathing had a thickness of 0.158 cm. The test conditions are given in Table VIII.

The effectiveness of aluminum and type 304 stainless steel as hydrogen barriers in organic exposed to a neutron flux was also tested. Some clad specimens together with unclad Zircaloy-2 were placed at "in flux" positions of the Z-7 organic loop. A similar group of specimens was placed in an "out-of-flux" section of the loop having the same temperature as seen by the in-flux specimens. The test conditions are given in Table IX.

III.3 Results:

The results of the three "out-of-pile" tests are given in Table VIII. The results of the "in-pile" test are given in Table IX.

III.4 Discussion of Results:

The conditions of test I (400°C for 2 weeks) were such that all specimens picked up relatively little hydrogen. The autoclaved and aluminum clad specimens picked up about 10 ppm hydrogen which is 30% of that picked up by the abraded Zircaloy-2. The autoclaved specimens heated in helium seem to have picked up considerably more hydrogen

than those autoclaved only. The 6 hours at 600°C probably caused considerable diffusion of oxygen into the sub-strate metal resulting in a decreased thickness of the protective oxide film.

Test II was run at 400°C for one week followed by another week at 450°C. These conditions are much more severe than those of Test I. Not only is the temperature higher, but the evolution rate of hydrogen from the dissociation of the organic is increased by orders of magnitude. This is very obvious from the amount of hydrogen picked up by the abraded Zircaloy-2: up to 20,000 ppm as compared to 39 ppm in Test I. It is apparent that the aluminum cladding offers a very high degree of protection, reducing the hydrogen pcik-up by a factor of about 200. The various treatments of the aluminum surface seem to have negligible effect. The 304 stainless steel caldding, although offering some protection, is not comparable to aluminum. Unfortunately no autoclaved specimens were included in the test.

Test III, carried out in an "out-of-pile" loop, yields no new information. A rough comparison of the conditions of this test with those of Test I show the results of the two tests to be in good agreement with one another.

The results of the tests carried out in the "in-pile" organic loop are given in Table IX. The most significant of these results is that certain specimens picked up very little hydrogen under conditions of temperature and hydrogen pressure very nearly those which would exist in an organic-cooled reactor.

Several of the aluminum clad specimens picked up less than 1 ppm hydrogen. This is in close agreement with a theoretical value (~ 2 ppm) based on the known permeability of aluminum to hydrogen together with a rough average of loop conditions. Because of the relatively small size of the aluminum clad specimens used in the in-pile test, great difficulty was experienced in obtaining a sound weld. It is therefore very probable that the higher rate of hydrogen pick-up shown by some of these specimens was due to pin-holes in the weld, enabling hydrogen to come in direct contact with the Zircaloy.

The two "in-flux" abraded specimens also picked up less than 1 ppm hydrogen, whereas the corresponding "out-of-flux" specimen picked up 91 ppm hydrogen. The moisture content of the organic very likely reduced the hydrogen pick-up by maintaining a protective oxide on the Zircaloy surface, but this should also affect the "out-of-flux" specimen to the same extent. There is no obvious reason why a neutron flux should lead to a decrease in hydrogen pick-up, so that the discrepancy as yet remains unexplained. More specimens for analyses would have been highly desirable.

As in the "out-of-pile" tests, the stainless steel clad specimens offered negligible protection against hydrogen pick-up.

IV. CONCLUSIONS

There tests have shown that some zirconium alloys may be exposed to Santowax OM plus 30% OMRE high boilers in the neighborhood of the operating temperature (375°C) of the pressure tubes of the proposed Canadian organic-cooled reactor for 97 days without appreciable hydriding.

The ATR alloy showed somewhat less hydriding than Zircaloy-2 after a 24 day exposure in organic at 260°C

Type 304 stainless steel is not an effective barrier to hydrogen.

Aluminum is effective in greatly reducing hydrogen pick-up of Zircaloy-2 in Santowax OM plus 30% OMRE high boilers.

A fast neutron flux of about 1×10^{13} n/cm^2 sec has no apparent effect on hydrogen pick-up.

V. SUMMARY

The mechanism of the corrosion of zirconium alloys in organic coolants is not clear. Under certain conditions the corrosion of specimens can be extremely severe, and large weight changes are observed. A small part of the weight change can be attributed to hydriding of the sample, but the remainder must be caused by the formation of an unidentified corrosion product. Where water contents are high, there may be some justification in referring to the corrosion product as "oxide".

High partial pressures of hydrogen result in severe hydriding and rapid corrosion of all alloys. Water contents in the range of 100 - 200 ppm are not harmful and may be beneficial, especially when hydrogen pick-up is being impeded by a protective oxide film.

It appears possible that, by suitable choice of alloy and conditions, zirconium alloys may find application in some organic reactors. No effects of reactor radiation have been observed which might be interpreted as indicating an enhanced corrosion and hydriding of zirconium alloys in irradiated organic coolants, though it must be recognized that neutron fluxes so far employed have been very low compared with the intensities in proposed power reactors. The results to date, although scattered, are sufficiently encouraging to warrant further investigation under carefully controlled conditions.

It is probable that aluminum, because of its stable oxide, may be useful as a hydrogen barrier at temperatures higher than those at which zirconium-oxide is adequate. However, the technical problems involved in applying aluminum cladding to zirconium alloys have not been solved and may have no simple solution.

VI. ACKNOWLEDGEMENTS

The design of the samples, holder and disassembly jig used in this experiment was by R. L. Beck. The analytical results listed in the text were obtained by Development Chemistry Branch and Reactor Loops Branch.

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Table I

Hydriding of Zirconium in MIPB 6 days at 400°C (Ref. 3)

H_2 partial pressure over MIPB	H_2O content of MIPB	ppm H picked up in Zr
0.11 atm	60 ppm	700 - 8700
0.34	100	460 - 630
0.34	150	80 - 130

Table II

Comparison of Weight Gain and Hydrogen Pick-up of Zirconium in Organics

Material	Coolant	°C	Hours	Atm	Cover gas	ppm H_2O	Wt gain mg/dm^2	ppm H picked up	Reference
Zirconium	Terphenyl	400	720	1.2	O_2	-	45	195	Davies (2)
"	"	"	"	"	"	-	43	205	Samples 6.35 x 0.64 x 0.064 cm
"	"	"	"	"	"	-	-70	435	Initial H conc : 5 ppm
Zirconium	IPB	371	500		(H_2)*	135	4	∿ 0	Boyd et al (5)
	"	371	200	20	H_2	175	22	∿100	Samples 5.1 x 1.27 x 0.16 cm
	"	"	500	"	"	"	104	960	Initial H conc : 35 - 50 ppm
		400	96		(H_2)	-	160	1880	
	Biphenyl	427	504		(H_2)	420	86	∿ 0	
	"	"	1000		"	"	128	∿ 230	
	"	"	1500		"	"	640	7140	
	"	"	"		"	<100	26	∿ 0	
Zircaloy-2	Biphenyl	427	1500		(H_2)	420	109	2090	
	"	"	"		"	<100	36	∿ 0	
Zircaloy-3A	Biphenyl	427	1000		(H_2)	420	137	∿ 230	
	"	"	1500		"	"	706	3380	

* (H_2). This represents hydrogen accumulated during run from decomposition of coolant.

Table III

Weight Gain and H Pick-up of S R E Zirconium
NAA-20 Loop Data (MTR) Kline, Gioseffi and Bley (6)

Coolant:		o, m-terphenyl
Bulk temperature:		260° - 343°C
System pressure:		13 - 20 atm N_2
Linear flow past specimens:		0.15 m/sec
Time:		324 h
Integrated fast flux:		1 x 10^{18} n/cm^2
	In flux	Out of flux
Weight gain	+ 53 ± 15 mg/dm^2	+ 24 ± 4 mg/dm^2
H pick-up	~ 80 ppm	~ 80 ppm

Table IV

Results of In-and Out-of-Reactor Tests on Zircaloy-2
in Organic Loop, NRX (7)

Sample size:	15 cm x 2.54 cm x 0.16 cm
Initial H content of material	40 ppm
Coolant:	Monoisopropylbiphenyl
H_2 Content:	20 cc/Kgm
H_2O Content:	100 - 160 ppm (nominal)
Ave fast flux:	1.1×10^{12} n/cm^2 sec
Integrated fast flux:	10^{19} n/cm^2

Coolant bulk temperatures and times of exposures of samples:

a) Out-of-reactor samples: 32 days at 150°C
    70 days at 232°
    24 days at 254°

b) In-reactor samples: 70 days at 232°C
    24 days at 254°

Sample Surface Preparation	Position	H Content at end of test, ppm
Pickled + 3 weeks in water at 360°C	Out-of-reactor	61, 61, 61
"	In-reactor, upstream from fuel	56, 59
"	In-reactor, downstream from fuel	61, 62, 66
As-pickled	In-reactor, upstream from fuel	35, 34

Table V

Composition of the Alloys

Element	Zircaloy-2		ATR Alloy	
	Malloy-Sharon Ingot 28094	Wah Chang Heat OMC-486-D6	A. I. John Thompson Heat ZI-9004	
	Alloy Elements (wt %)		Alloy Elements (wt %)	
Fe	0.123	0.11	Cu	0.45
Cr	0.079	0.12	Mo	0.54
Ni	0.053	0.06	Zr	Balance
Sn	1.39	1.55		
Zr	Balance	Balance		
	Impurities (ppm)		Impurities (ppm)	
Cu	25	< 20	Sn	20
Mo	< 20	10	Fe	260
Al	33	33	Al	120
Ti	< 20	< 20	Ti	< 30
Hf	< 45	120	Hf	42
B	0.25	< 0.2	O	1200
Co	< 5	< 5	N	156
Cd	< 0.2	< 0.3	H	10
Mn	< 10	12		
W	< 20	< 40		
Mg	< 15	< 10		
Pb	< 15	10		
Si	< 30	50		
V	< 20	< 20		
C	167			
N	53	58		
H	25	6		

Note: Hydrogen removed by vacuum anneal prior to tests.

Table VI

Summary of Weight Change Data

Sample	Initial Weight g	Wt. Gain after Autoclave mg/dm^2	Wt. Change in loop Exposure mg/dm^2
Zircaloy-2			
Z-1	2.5272	11	8
-2	2.5353	21	0
-3	2.5493	21	-1
-4	2.5401	21	0
-6	2.5450	not autoclaved	0
-7	2.5396	"	4
-8	2.5363	"	0
-9	2.5131	"	0
A T R Alloy			
A-1	2.5865	57	-12
-2	2.5466	56	-12
-3	2.5600	67	-24
-7	2.5792	not autoclaved	1
-8	2.5284	"	7
-9	2.5201	"	4
-10	2.5849	"	2

Table VII

Summary of Essential Results on H Pick-up

Zircaloy-2

Exposed in as-pickled condition

Initial H content	-	2 - 3 ppm
Final H content	-	7 ± 3
Average pick-up	-	4 - 5

Exposed in autoclaved condition

Initial H content	-	38 ± 8 ppm
Final H content	-	51 ± 7
Average pick-up	-	13

A T R Alloy

Exposed in as-pickled condition

Initial H content	-	< 2 ppm
Final H content	-	4 ± 4
Average pick-up	-	∿ 2

Exposed in autoclaved condition

Initial H content	-	57 ± 3 ppm
Final H content	-	57 ± 10
Average pick-up	-	0

Table VIII

Out-of-Pile Hydrogen Pick-up Results

	Test Conditions		
	Test I	Test II	Test III
Test facility:	S.S. bomb	S.S. bomb	out-of-pile loop
Coolant:	Santowax OM	Santowax OM	Santowax OM + 30% OMRE high boilers
Duration and temperature:	14 d at $400^{\circ}C$	7 d at $400^{\circ}C$ + 7 d at $450^{\circ}C$	7 d at $365^{\circ}C$
H_2O content of coolant:	154 ppm	154 ppm	260 ppm
Estimated average H_2 pressure:	~ 0.5 atm	$decomp^n$ H_2	1 atm
Coolant velocity:	-	-	270 cm/sec

	Hydrogen Picked up (ppm)		
Specimen Treatment	Test I	Test II	Test III
Abraded Zr-2	28, 39	16,500 – 22,000(3)	62 – 92 (6)
Heated in He	33, 40		74 – 93 (4)
Autoclaved	9, 13*		10 – 38* (4)
Autoclaved + He	29, 38*		10 – 26* (4)
Heated in air			
S.S. clad		2,300 – 6,000 (7)	
Al clad	8, 11	81 – 113 (5)	
Al + air heated		81 – 88 (3)	
Al + air heated + He		60 – 71 (5)	

* Corrected for hydrogen pick-up during autoclaving
( ) Number of analyses made

Table IX

In-Pile Hydrogen Pick-up Results

	Test Conditions
Coolant:	Santowax OM + 30% OMRE high boilers
Duration of run:	97 days of which 76 days were at power
Coolant temp. at specimen:	290 - 365°C
Coolant velocity over specimen:	negligible
H_2O content of coolant:	113 - 157 ppm
H_2 partial pressure:	0.034 - 0.57 atm
Ave fast flux (> 500 ev):	~ 1 x 10^{13} n/cm^2 sec
Integrated fast flux:	~ 6.5 x 10^{19} n/cm^2

Cladding	Position	H_2 Pick-up (ppm)
Abraded Zr-2	out-of-flux	91
Al (0.079 cm)	"	< 1
S.S. (0.079 cm)	"	371
Abraded Zr-2	in-flux	< 1, < 1
Al (0.079 cm)	"	< 1, 130, 166
Al (). 158 cm)	"	< 1, 47
S.S. (0.079 cm)	"	96, 117, 258
S.S. (0.158 cm)	"	107, 207, 145

SUMMARY

PROGRESS & PROBLEMS WITH FUELS & FUEL CLADDING

FOR ORGANIC REACTORS

by

H. Pearlman,
Atomics International

Experience at Atomics International with organic reactor
fuel elements has been almost entirely with aluminum-clad U-Mo
alloys, and APM-clad UO_2 pellets.

Fabrication of U-3.5Mo-0.1Al fuel for the first core
loading of the Piqua, Ohio OMR is approaching completion. The fuel
is made by direct casting. It is metallurgically bonded through
a layer of electroplated nickel to finned Type 1100 aluminum cladding.
Piqua operating conditions include a maximum (hot channel) cladding
surface temperature of 718°F (380°C). Irradiations of model elements
of this type have been made over a temperature range from 650° to
750°F, to burnups near 3000 Mwd/MTU. It is concluded that Piqua
fuel elements will be sufficiently stable dimensionally, and will
resist adequately the aluminum--fuel alloy interdiffusion reaction,
to attain the design burnup of 3000 Mwd/MTU average.

More advanced OMR designs require metal fuel burnup in
the range from 10,000 to 20,000 Mwd/MTU, at peak surface temperature
to 750°F (400°C). U-10Mo alloy offers promise of sufficient stability
under irradiation to permit such burnups. Data are available from
irradiations of this alloy, mainly at higher temperatures, which show
adequate dimensional stability at high burnup. However, there are
as yet no experimental results under exact OMR conditions of geometrical
configuration, temperature profile and burnup rate. AI has recently
completed a capsule irradiation of multiple, small plate samples of
14 different U-Mo alloys, ranging from U-3.5Mo to U-10Mo, and including
ternary and quaternary compositions. Temperatures were between 750°F
and 950°F (400°C and 510°C), and peak burnup was near 15,000 Mwd/MTU.
Fission rates in this experiment were somewhat higher than expected
in advanced design, metal-fueled OMR's, but it is expected that this
experiment will furnish valuable information on the behavior of the
fuel alloys. Hot cell examination has just begun. It appears worth-
while to expand the metal fuel development work to other compositions.

The relatively rapid diffusion of aluminum into nickel at
advanced design OMR temperatures, together with the long in-core
residence times required to achieve the required high burnup, make
it desirable to have a better barrier material than nickel. Experi-
ments have shown that both Ti and Nb are superior, but Nb is clearly

best. Isothermal soak tests have been made with metallurgically bonded compacts of U-Mo to Nb to aluminum. The initial high quality of bond and barrier effectiveness are retained at least to 10,000 hours at 900°F. Methods for effective application of Nb to U-Mo alloy are still under development. Deposition of Nb vapor on the fuel alloy, in vacuum, from metal superheated by an electron beam, appears to be the most practical production technique.

Fabrication procedures utilizing Nb are not yet sufficiently advanced to build model elements for irradiation testing. However, a U-10Mo, plate-type element with Ni barrier is now in OMRE, awaiting irradiation. Design burnup is 10,000 Mwd/MTU, at peak surface temperature of 750°F.

For higher performance, at peak surface temperature of 850°F and peak burnup of 35,000 Mwd/MTU, UO_2-APM fuel elements are promising. Compatibility of UO_2 and APM (Type M-257) has been established at least to 5000 hours at 1100°F (595°C).

Coking of the organic coolant on contact with hot UO_2, through a defect in the cladding, is a possibility. Experiments, both under irradiation and out-of-pile, have shown that growth of the coke deposit between the UO_2 and the cladding can occur. Under certain conditions, the cladding may crack because of mechanical strain. This is a slow process, not expected to cause a significant operational problem. More experiments are needed to define it.

Mechanical properties of sheet, bar, and tubing have been determined for M-257, M-486, SAP-930 and XAP-001 over a range of temperatures. M-257 appears to be the most satisfactory; but XAP-001, an equivalent, chemically "cleaner" composition, may prove preferable. There is still a general requirement for greater consistency in the quality of billet stock and mill products.

Satisfactory finned APM tubing for cladding, and coolant guide boxes and other structural shapes, can be made by extrusion of billet stock. Tube closures have been successfully made by eutectic bonding and flash welding. Other joints can be made by these methods, and by cold pressure welding.

Fabrication and non-destructive test techniques have been developed sufficiently well to permit construction of a number of APM-UO_2 model elements. Four are now in the OMRE core, awaiting irradiation. They consist of an array of 24 or 25 fueled tubes, made of several different APM compositions, with boxes made of a variety of materials. Design burnups are from 8000 to 30,000 Mwd/MTU, at peak surface temperatures from 750°F (400°C) to 850°F (455°C). Progress in irradiation testing has been hampered by limited facilities.

REPORT ON THE STANDARDS AND ADVISORY COMMITTEE
OF THE U.S. ORGANIC REACTOR WORKING GROUP
Robert O. Bolt

In 1960 the need became apparent for give and take discussions
of research in the Organic Reactor Program. This led to the
formation of the Organic Reactor Working Group, made up of all
the organizations in the U.S. engaged in work in this field.
AECL personnel have also participated from early in the program.
Meetings are held every three to four months at various sites
at which organic reactor work is being conducted. There has
been a wide interest in these sessions in spite of their
frequency, and 50-100 people have attended each meeting.

This large attendance did not permit detailed discussions of nar-
row topics, such as test standards in coolant technology. Conse-
quently, late in 1961 the Standards and Advisory Committee was
formed from the following major participants in the Working Group:

> Atomic Energy Commission (ex officio)
> Atomic Energy of Canada, Ltd.
> Atomics International
> Brookhaven National Laboratory
> California Research Corporation
> Massachusetts Institute of Technology
> Monsanto Research Corporation
> Phillips Petroleum Company

The Committee functions in a general advisory capacity on coolant
technology to the Working Group. Its initial assignment is on
standardization of test procedures and equipment.

In its initial work, the Committee has reviewed pertinent test
methods and equipment and data relative to test performance.
Based on this review, standardized versions are agreed upon and
proposed for use by the Working Group organizations. The work
has been largely carried on by the permanent members of
the Committee, but from time to time others have been invited to
participate as the subjects under discussion dictate. Meetings
have generally been held in conjunction with Working Group
sessions, but special sessions have also been called.

Committee Activity

Table I summarizes the general areas chosen for committee activity.
Some of these have received attention, as indicated, in the first
of the committee working sessions.

Table I

Standards Areas Chosen for Committee Activity

        Coolant Characterization*
        Fouling Tests*
        Standard or Calibration Samples*
        Radiation Measurements*
        Sampling Procedures*
        Gas Analyses
        Corrosion Tests
        Coolant Purification
        Nomenclature

        * Committee action

Table II gives a further breakdown of the coolant characterization area. Topics that have received committee attention are also indicated.

Table II

Areas for Committee Activity on Coolant Characterization Methods

        Iron,* Other Inorganics Content
        Ash Content*
        Water Content*
        Oxygen Content*
        Particle Size
        High Boilers (small and large-
          scale tests)
        Low Boilers
        Molecular Weight
        Structure
          NMR, IR, UV, MS, GC
          o-, m-, p-terphenyl analysis
          C-H ratio
        Viscosity
        Density
        Vapor Pressure

        * Committee action

The Committee has recommended for general use a method of iron determination by X-ray emission spectroscopy. Work is going forward under committee auspices on methods of determining ash content, particularly those that utilize small samples and require short times. However, no method is sufficiently developed as yet to be recommended for general use. A similar

situation exists with respect to committee work on water content and oxygen content. With the latter, research on methods involving neutron activation is proceeding. Work is barely under way on means of determining particle size.

Table III summarizes the topics for committee work on <u>fouling tests</u>. A pyrolytic capsule test procedure has been recommended for general use. However, research on improved procedures is also being pushed vigorously. It is hoped to increase the reliability of the existing method by providing equipment with which better cleanliness is possible. The effect of agitation is also being investigated.

<u>Table III</u>

Areas for Committee Activity
on Fouling Tests

Exchange of Procedures*
Improved Reliability*
Removal of Film* and Analysis
Temperature Measurement and Control*
Mechanical Agitation*
Small-Scale Test*
Effect of Irradiation During Test*
Exchange of Test Samples*
In-loop (Reactor Tests)

* Committee action

The existing method requires 250-300 ml of test fluid. There is a need for results with much smaller amounts, and work is under way on a method that uses less than 10 ml of fluid. Effects of irradiation during the pyrolytic test are also being studied with both this miniature and the larger-scale equipment.

In the development of tests of this sort, the availability of standard samples is of utmost importance. Samples irradiated in the OMRE are being set aside for this purpose along with samples irradiated in a linear accelerator. These are available through the Committee as required.

Table IV describes committee activity on the remaining <u>miscellaneous items</u> listed in Table I. For example, 4500 pounds of standard Santowax OMP has been put aside for the use of the Working Group members. This will assure an adequate quantity for various experiments. Sampling procedures, particularly those involving large samples, have also received attention.

The important area of radiation measurements is under study by the Committee. It is the consensus that calorimetry involving various absorbers has the best prospects for yielding a

standardized procedure. The conversion of data from calorimetry into information meaningful in the Organic Program is also of vital interest. Recommendations in this area are expected from committee activity in a short while.

Table IV

Areas for Committee Activity on
Miscellaneous Items

Standard or Calibration Samples
   Pure Compounds for Instrument Calibration
   Materials for Test Standardization*
Sampling Procedures
   Large Scale*
   Small Scale
Radiation Measurements
   Standard Methods*
   Use of Data in Organic Systems*
   Computer Treatment of Data

   * Committee action

Future Work

The Committee is now well established and at work in the standards area. A steady flow of procedures derived from this work is expected. These will be of real importance in research of the Organic Reactor Program. As the effort on standards becomes less intense, it is expected that the Committee will be utilized in other capacities.

Some Recent Developments in Organic Coolant Studies

By
R. H. J. Gercke

Atomics International
A Division of North American Aviation, Inc.

Summary

This presentation will cover recent progress in our studies on
coolant decomposition, and on surface film formation including recent
OMRE experience. The decomposition rate of the organic fluid is of
major interest only in reactors using the organic as both moderator
and coolant, as the makeup rate in D_2O moderated reactors represents
only a minor contribution to the cost of electric power. In large
organic moderated and cooled central power stations, the makeup cost
of the organic coolant may represent as much as 10% of the cost of
power generation and therefore contribute significantly to the economics
of this type of reactor.

While the effect of electrons and gamma rays on the decomposition of
the terphenyls has been well established for quite a few years, con-
siderable uncertainty has continued to exist with respect to damage
from thermalization of fast neutrons. British experimenters at Harwell
were the first to point out that for the same amount of energy absorbed,
fast neutrons did considerably more damage to the terphenyls than either
electrons or gammas. We have recently conducted extensive irradiations
on terphenyl coolant at the Curtiss Wright Research Reactor and the
Oak Ridge X-10 Reactor. These experiments have confirmed the British
results. Our results show that at 600°F the decomposition rate of the
terphenyls under exposure to fast neutrons is approximately four to
five times greater than that under exposure to electrons or gammas.
The data are still too limited to allow a reliable definition of the
temperature dependence of this "fast neutron effect".

Improved calculations of ionizing energy absorption in the coolant
of organic moderated and cooled reactors show that inelastic scattering

of neutrons in the fuel elements and minimization of the volume of
organic present in the core decrease the amount of energy absorbed in
the terphenyls to less than 3% of the total reactor power (with a
fast neutron contribution of approximately 55% of this fraction).
The resulting coolant makeup rate at a high boiler concentration of
30% will generally (depending upon the reactor design and operating
conditions) be around 4.18 lb/Mwhe, which at a price of 17¢/lb for
unirradiated terphenyl, corresponds to a makeup cost of approximately
0.7 mills/Kwhe. Coolant makeup rates have recently been calculated for
a first generation 300 Mwe deuterium moderated-organic cooled reactor.
The makeup rate is 171 lbs/hr at full power which is equivalent to
0.6 lb/Mwhe or 0.09 mills/Kwhe.

The problem of fouling or formation of deposits at the surface of
the fuel elements, with consequent possibility of insulation of the
cladding surfaces and plugging of the coolant channels, is a fundamental
one as it directly affects the feasibility of the concept of an organic
cooled reactor. Considerable information has been developed in this
area from the experience of the operation of the OMRE as well as the
operation of the NRX X-7 in-pile loop and of various out-of-pile loops
and experiments. As a result, various theories have been proposed to
explain the mechanism of film formation. These are all based on the
effects of the presence of impurities in the coolant (corrosion
products, very high molecular weight decomposition products, oxygenated
hydrocarbons, oxygen and water). All the evidence now shows that good
control of the cleanliness of the coolant will result in acceptably low
film formation rates in organic reactors. The major problem in film
formation, therefore, appears to be the engineering one of controlling
the purity of the coolant.

Initial irradiation of the Third Core of the OMRE was carried out
through 282 MWD of reactor operation. Because of pressing schedule
problems, power operation had to be initiated prior to completion of
installation and checkout of the coolant purification equipment
originally planned for this operation. Nevertheless, the purity of
the coolant was improved by more than a factor of 10 over that of the
coolant used during the operation of the Second Core. This was
achieved by increased processing through the distillation system (at

a rate of 1 gpm or one coolant inventory in 4 days. The results proved highly encouraging. While appreciable deposits were observed at places of localized flow obstructions (straps holding fuel plate bundles together, loose thermocouple in a coolant channel, thermocouple junctions on outside of surfaces of fuel plates), the deposits everywhere else, where no disturbance to the flow or flow stagnation could take place, generally averaged 0.080 to 0.087 mills with the maximum thickness being 0.17 mills. As observed in all previous cases of fouling, the deposits had a high iron content, with from 25 to 70% ash. It is to be noted that the film formation rate observed at OMRE (0.0014 mil/operating day) would result in a fuel element cladding surface temperature rise of 0.07°F per day at a heat flux of 300,000 BTU/ft^2-hr (using a value for k of 0.5 BTU/hr-ft°F). Thus, one years operation would result in the deposition of 0.50 mills film with a Δt of 25°F. To further reduce the film formation rate during the next period of operation of the OMRE, additional coolant impurity removal equipment is being put into operation and the capacity of the distillation system is being increased. The operation of the reactor is scheduled to resume with circulation of the coolant through a 40 micron full flow and 3 micron bypass filters, as well as processing of a 1.9 gpm flow through a clay adsorption column and a 3 gpm flow through the distillation system. This increased processing rate is intended not only to decrease the steady state concentration of impurities in the coolant, but also to eliminate the potential effects of sudden releases into the coolant under thermal or hydraulic cycling of impurities accumulated over the years at various stagnant points in the piping system.

Operation of the OMRE will be resumed in the latter half of June 1962. This next period of operation of OMRE is expected to further demonstrate that acceptably low film formation rates can be achieved during operation of an organic reactor.

SEARCH FOR NEW COOLANTS AND RECLAMATION OF COOLANTS;

THE U. S. PROGRAM

J. C. Hillyer
Phillips Petroleum Company
Bartlesville, Oklahoma

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A B S T R A C T

The objective of the AEC research program on new coolants and coolant reclamation is to reduce the cost of power generation from an organic cooled reactor. With the present Santowax coolant, replacement of fluid destroyed by radiation is a major factor in the operating cost; a more stable and/or a cheaper coolant would be advantageous. Similarly, reclamation of usable coolant from the presently discarded high boiling degradation products could offer an economic advantage. Research programs designed to achieve these objectives are in progress under AEC contracts at Atomics International, Monsanto Research Corporation, and Phillips Petroleum Company.

A carefully controlled search for better coolants is under way at Phillips Bartlesville research laboratories, built on the extensive surveys and evaluations which have been done earlier, and following in its development the development of basic knowledge and theories of radiolysis resulting from our own and other work in this area. The possibility of producing an acceptable coolant from available very low cost by-product streams from the petroleum or coal-tar industries is being resurveyed in view of greatly increased knowledge of their composition and new processing methods in these industries.

The program of synthesis and evaluation of new compounds as candidate coolants is now actively in progress. Areas selected for initial investigation include several types of nitrogen heterocycles, which present knowledge and a survey of all earlier work indicate have been inadequately investigated. In particular, pyridine ring-containing compounds, triazoles, and thiophene and thiazole types are being evaluated. Partially hydrogenated condensed ring aromatics have in preliminary tests shown marked stability to electron irradiation. This very promising area is receiving intense study, and in-pile irradiation tests will be included.

Another approach to more stable coolants under investigation comprises the study of mixed systems in which a minor amount of an added compound serves to stabilize terphenyl coolant. Several types of compounds have been found in which 1 to 3 per cent added to Santowax reduces the total radiolytic damage by 15 to 25 per cent.

Previous surveys of petroleum refinery streams by California Research Corporation and Atomics International indicated some possibilities in low sulfur, high boiling aromatic oils. Modern refining techniques involving catalytic cracking, solvent extraction and preparation of high aromatic feedstocks for furnace black production have made available a wealth of likely candidate materials. Current investigations at Phillips have revealed several typical streams which, on dealkylation, dewaxing, desulfurizing, etc., have exhibited stability toward electron irradiation actually exceeding that of terphenyls and which are thermally stable up to 700 F. Further processing it is hoped will result in extending thermal stability above 725 F and in reduction in the amount of gas generated on radiolysis. New analytical techniques are revealing much needed information on the compositions of these fractions and make it possible to follow closely and accurately the effects of our processing operations.

The objective of the research program which the Commission has sponsored in reclamation of used coolant is, like the new coolant program, to reduce reactor operating costs due to coolant replacement. Additional inducements to developing a successful reclamation process lie in saving the present cost of waste disposal and solution of the bulk handling problems of waste and replacement coolant. This program has resulted in the demonstration of the technical feasibility of two or three reclamation processes; these appear to be economically attractive also, and are being critically compared.

Reclamation methods initially studied were applied to high boilers separated from used coolant by a distillation; these comprised all materials boiling above p-terphenyl. Chemical methods studied have included hydrocracking, partial reduction, and disproportionation reactions; the physical methods are solvent extraction and severe distillation. More recently both solvent extraction and hydrocracking have been applied to the total coolant with outstanding success. Saving the investment and operating costs of the initial distillation step would be most attractive.

Extension of the range of material recovered on distillation of coolant to include compounds boiling above p-terphenyl is the simplest means of reducing the loss of coolant. Studies by Atomics International on adaptation of the usual units to more severe "deep distillation" have indicated that about 30-50 per cent of the "high boiler" can be recovered as material with suitable properties for blending into terphenyl coolants and satisfactory radiolytic stability. The economic advantages are obvious.

Molecular distillation has offered certain difficulties in practice, but other short-path high vacuum apparatus has been successfully operated. Research still in progress at Monsanto Research Corporation on use of a commercial wiped-film apparatus has shown recovery of about 25 per cent of the high boiler feed as an extremely stable coolant at about 8.5 to 9.0 cents per pound including amortization of the investment.

A process perhaps basically more promising is one of solvent treatment in which the segregation can involve molecular types as well as molecular weights. Monsanto has shown that the addition of aliphatic hydrocarbons to aromatic solvents for high boiler results in segregation of a high boiling, tarry fraction. From a typical VM&P naphtha containing 70 per cent paraffins and naphthenes, 75 per cent of high boiler is reclaimed as a usable fraction, m.w. about 400, of low fouling potential and low oxygen content.

The chemical method receiving the most attention for high boiler reclamation is hydrocracking. The large polymeric polyphenyl molecules are broken down into terphenyl-range material by catalytic treatment under hydrogen pressure. Two approaches have been used: single stage operation in which scission between rings and hydrogenation are accomplished simultaneously; and a two stage operation in which the high boiler is first subjected to mild catalytic hydrogenation followed by cracking in a separate step. In the single stage operation, very successful results have been demonstrated by Phillips Petroleum Company using relatively low conversion, recycling the residues to extinction. Very long operating cycles are achieved, with low coke deposition, low hydrogen consumption and conversion of over 90 per cent of the high boiler to coolant. The recovered material contains a small amount of alkyl poly-phenyls as well as terphenyls, has thermal stability and radiolytic polymer formation

better than Santowax and tolerable gas generation. Monsanto Research Corporation
has shown that hydrocracking 8 to 16 per cent hydrogenated high boiler at 40-50
per cent conversion gives yields of 60-80 per cent of a good coolant consisting
almost entirely of alkyl biphenyl and alkyl terphenyls. Economics for these
hydrocracking operations are favorable at recoveries above 75-80 per cent. Coolant
can be reclaimed in a commercial size unit at about 7.5 cents per pound including
amortization.

A development most favorable to the economics of these recovery processes
is the discovery that they can be applied to the high boiler as it exists in the
coolant, without the need for preliminary segregation by distillation. Hydrocracking
can convert 95 per cent of the high boiler content of used coolant in a single pass,
with 95 per cent total recovery of a clean, stable coolant. Solvent redistribution
is also successfully being applied to total coolant. In this process the terphenyl
content of the coolant probably supplements the aromatic components of the mixed
solvent. With solvents of 20 to 30 per cent aromatic content, Monsanto estimates
that recovery of a good coolant fraction can be made on coolants containing 20 per
cent or more of high boiler for 5 to 6 cents per pound including amortization.

INDUSTRIAL OIL COOLANTS (Phillips Petroleum Co.)

Oil Examined	Relative Radiolytic Yields	
	Polymer	Gas
1. "Alkyl Phenanthrene" Nuclear Reactor Coolant	1.11	9.3
2. Hydrodealkylated Cycle Oil- Fluid Cracking of Mid-Continent Crude	0.82	2.8
3. Re-extracted Heavy Aromatic Oil (Carbon Black Feed)	0.86	7.3
4. Phenanthrene Cut from Coal Tar "Anthracene Oil"	0.91	1.2

PYROLYSIS OF INDUSTRIAL OIL COOLANTS
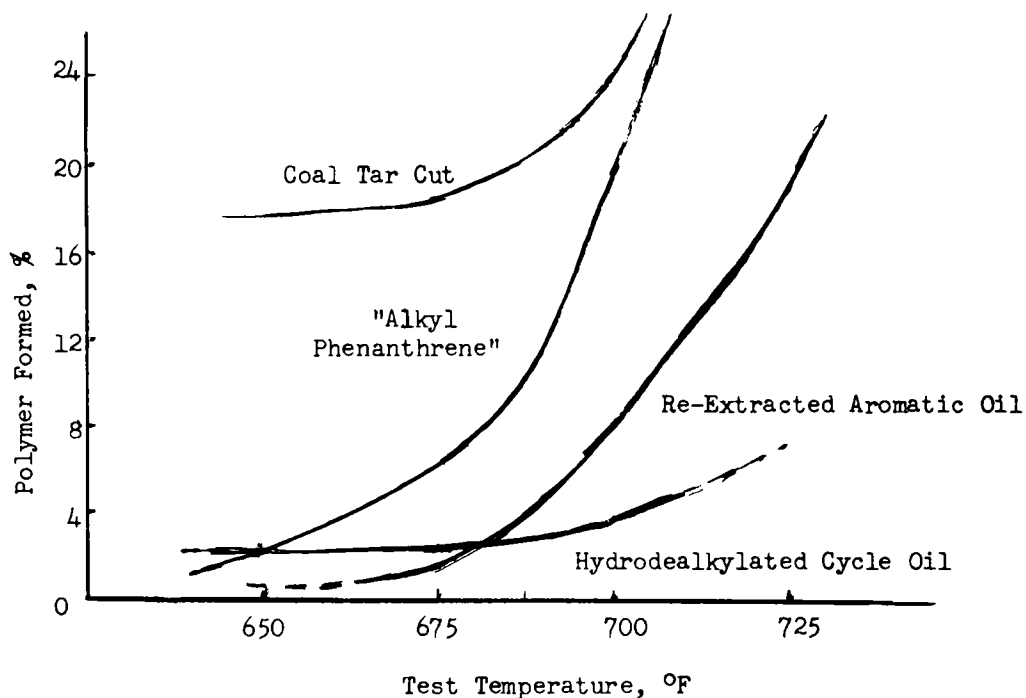
WIPED FILM DISTILLATION (Monsanto Research Corp.)

Pressure - 0.39 mm Hg   Still Temperature - 323 C

	H.B.	Distillate	Residue
Wt. % Yield	-	27	60
No. Avg. Mol. Wt.	550	343	749
C/H Ratio	1.40	1.38	1.49
PCFT, mgm Film	-	6.2	47.5
Thermal Decomp. Temp., °C	360	402	330
ϕ_3 Content, Wt. %	10	-	None

"DEEP DISTILLATION" (Atomics International)

Core III HB   5 mm Hg

Cut No.	O.H. Temp. (°F)	Wt. % In Fract.	Wt. % Cum.	Avg. Mol. Wt.	PCFT (mgm Film)
1	426	5	5	234	-
2	426	5	10	234	
3	426-445	5	15	235	?
4	445-477	5	20	269	
5	477-567	5	25	293	
6	567-621	5	30	328	2.4
7	621-660	5	35	427	
8	660-699	15	50	496	2.0
Feed	-	100	-	405	50.4

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RECLAMATION OF DAMAGED COOLANT BY CATALYTIC HYDROCRACKING (Phillips Petroleum Co.)

Process: Low conversion with distillation and recycle of unconverted high boiler

Catalysts: $CoMoO_4$-Al_2O_3, NiO-Al_2O_3, Pt-Al_2O_3, CoO-V_2O_5-Al_2O_3

Conditions: 850-950 F, 1000 psig, 1.0 LHSV, 20-50 mol hydrogen per mol high boiler

Summary of Typical Results at Steady State Catalyst Activity:

Conversion of High Boiler, Wt. %	25
Product Recovery, Wt. %	96
Coolant Yield Per Pass, Wt. %	23
Coke, Wt. % of High Boiler Charged	0.5
Hydrogen Consumption, Mol H2/Mol HB	3.3
Process Cycle Length, Hours	50-100
Initial Investment, Dollars	985,000
Reclaimed Coolant Cost, Cents/Pound	7

EVALUATION OF RECLAIMED COOLANT FROM HYDROCRACKING OMRE HIGH BOILER

	Distilled Reclaimed Coolant	Santowax-OMP
Melting Range	200-250	300-350
Viscosity at 260 F, Centistokes	2.1	2.1
Density at 260 F, g/cc	1.01	1.02
Avg. Molecular Weight	225	230
C/H Atomic Ratio	1.20	1.29
Composition, Wt. %		
Biphenyl	10	<1
Alkylpolyphenyls	22	-
Terphenyls	53	99-100
Triphenylene	10	-
Quaterphenyls	5	-
Radiolytic Stability*		
Polymer, Wt. %	16	24
Gas, cc/watt-hr	2.8	0.9

*Total dosage of 8×10^9 rads at 650 F under the Linac.

CATALYTIC HYDROCRACKING OF TOTAL COOLANT (Phillips Petroleum Co.)

Conditions: $CoMoO_4-Al_2O_3$, 1000 psig, 900 F, 1.0 LHSV, 20 mol hydrogen per mol coolant

		OMRE Core II Coolant
Conversion of High Boiler, Wt. %	96	
Coolant Yield, Wt. %	95	
Yields, Wt. % of Coolant Charged		
Lighter than Biphenyl	3	
Biphenyl	5	< 1
Alkylpolyphenyls	9	2
Terphenyls	74	71
Triphenylene	3	3
Quaterphenyls	3	< 1
High Boilers	1	23
Coke	2	
Coolant Molecular Weight (Avg.)	223	270
Radiolytic Stability*		
Polymer, Wt. %	21	24
Gas, cc/watt-hr	2.8	0.9

*Total dosage of 8×10^9 rads at 650 under the Linac.

TWO-STAGE HIGH BOILER HYDROCRACKING (Monsanto Research Corp.)

1st Stage - 15.8% reduction over nickel or copper chromite catalyst
2nd Stage - Cracking at 1500 psig, 500 C, 10/1 H_2/Feed ratio

Catalyst	Ni-Cr-Al$_2$O$_3$	MoS$_2$-Al$_2$O$_3$	Copper Chromite
Yield of Total Product	70	72	69
Yield of ϕ_2, ϕ_3	0.5	0.7	0
Yield of Rϕ_2, Rϕ_3	18	11	34
Yield of Rϕ, ϕH	7.4	8.3	-
Conversion, %	34	32	35
Product, M.W.	395	371	300
C/H	1.35	1.30	1.20
Decomp. Temp., oC	395	371	395

SOLVENT RECLAMATION OF SPENT COOLANT (Monsanto Research Corp.)

Aromatic/ Aliphatic Ratio	HB Conc., %	Wt. Treated per L Solvent, g	Yields, Wt. %	
			Reclaimed Coolant	High M.W. Fraction
30:70	10	360	96.0	1.5
30:70	30	449	94.0	4.8
20:80	30	270	91.5	6.3

Properties

	Coolant	Reclaimed Coolant
Mol. Wt.	275	260
C/H Ratio	1.35	1.28
Decomp. Temp., °C	416	452
PCFT, mgm	63	5

SOLVENT DISTRIBUTION METHOD (Monsanto Research Corp.)

VM&P Naphtha - (150°-200°) - 30% aromatic - 70% aliphatic

		H.B.
Temperature of Mix, °C	100	
Mixing Time, min.	2-5	
Concentration, g/liter solvent	333	
Solvent Recovery	90-95	
Wt. % Yield Reclaimed	75	
Reclaimed Product, M.W.	450	550
Viscosity, cp	5.0	10.7
Decomp. Temp., °C	370	360
PCFT, mgm	5	160

"Theoretical Studies on Liquid Coolants"

Department of Chemistry
University of Utah
Salt Lake City, Utah

I. THE ELECTRONIC STRUCTURES OF POLYPHENYLS AND HEXA-

FLUOROBENZENE COMPOUNDS (by T. Nakamura and T. Watanabe)

In connection with the radiation chemistry of polyphenyl compounds, the knowledge of the excited electronic states of these molecules is very important. Among them the "lower" excited states are due to the excitation of the pi-electrons and are observed as near ultraviolet spectra. Polyphenyls are rather large molecules and the theory of these excited states is accordingly complicated. For instance, a precise calculation of the excitation energies of polyphenyls seems to be impossible at present. However, the most important and interesting problem is to obtain a rather rough but physically reasonable idea or picture of the excited electronic structure. From this standpoint we are examining the lower excited states of polyphenyls.

In case of polyphenyls the pi-electronic structure of these molecules is closely related to those of their constituent parts, i.e., the benzene (phenyl) rings. Experimentally the existence of such relationships has been established by spectroscopic studies[1], etc. This fact suggests that a polyphenyl molecule is theoretically treated as a composite system consisting of benzene rings coupled together. We have recently discussed the (pi-electron) coupling of the benzene rings in polyphenyls in the ground state;[2] on the basis of a simple theoretical calculation it has been shown that the coupling is considerably weaker than was believed to be in the past or as calculated by a naive molecular orbital theory (the Hueckel theory). Similar situations involving the coupling between two conjugated double bonds in linear hydrocarbons have recently been discussed by many authors.[3]

This kind of "weak" coupling is expected also in the lower excited states. Actually Davidov[4] and Longuet-Higgins and Murrell[5] calculated the lower excited states of biphenyl using the weak coupling model with considerable success. The lower excited states of polyphenyls in general are being studied by us

on the basis of a similar model.

According to this model we start from tne pi-electron wave functions of the constituent benzene rings and calculate the interaction among the benzene rings as a perturbation. The perturbation is supposed to arise from two kinds of interactions; one is due to electron-electron repulsion and the other might be called the resonance integral interaction, which is an interaction through the core part or the (one-electron) Hamiltonian. The latter interaction results in the conjugation of the benzene rings, but is shown to be of minor importance compared with the electron-electron repulsion. Actually Davidov neglected this conjugation effect in his calculation.[4] However the conjugation has an interesting effect on the lower excited states; i.e., the excitation which corresponds to pi-electron transfer from one ring to another[5] can arise because of this conjugation. The magnitude of the resonance integral interaction is related to the co-planarity of phenyl molecules,[2] and therefore we are calculating the change of excitation energy and the oscillator strength with the twisting angle between two benzene rings. The comparison of the calculated results with spectroscopic observation will be useful for the determination of the configuration of polyphenyls.[2]

We are also interested in the physicochemical properties of hexafluorobenzene which is also used as an organic coolant. The electronic structure of fluorinated benzenes is most conveniently discussed in connection with that of benzene: one examines the change of the electronic structure or benzene when the hydrogen atoms in the benzene molecule are substituted by fluorine atoms. Two main effects of such a substitution on the pi-electronic structure are well-known: one is the mesomeric effect (i.e., the conjugation of the pi-electrons of substituent atoms with the aromatic pi-electrons) and the other the inductive effect. In the case of fluorine substitution, however,

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the latter effect is supposed to be predominant.[6] The inductive effect

in fluorinated benzenes has been discussed by Murrell and Longnet-Higgins.[6]

They calculated the change of the excitation energy due to the fluorine sub-

stitution by using the perturbation theory, and compared the calculated results

with Sponer's experimental observation.[7]

Among others, the hexafluorbenzene molecule have the same symmetry as

benzene, so that its pi-electronic structure resembles that of benzene. A few

calculations relating to the inductive effect in the hexafluorobenzene molecule

are in progress. Also we think that the theoretical study of the C-F bond of

this molecule is important in this connection. The experimental data such

as the dipole moments of fluorinated hydrocarbons, and nuclear magnetic

resonance data (chemical shift, spin coupling constants, etc.) of fluoro-

benzenes[8] are considered very important sources of information for theoretical

thinking.

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STUDIES ON PERFLUORO-PHENYLS

(by W. C. Lu)

The literature survey on perfluoro-benzene and perfluoro-polyphenyls (bi-phenyl and o, m, p-terphenyls) reveals that there is no physico-chemical study of the latter, whereas some references are found which deal with the properties of perfluoro-benzene. For example, Dibeler, Reese and Mohler[1] carried out an electron-impact experiment to study the ionization and dissociation energy of C_6F_6. The results are as follows: the ionization potential of C_6F_6 is greater by approximately 0.7 ev than that of benzene and the dissociation energy $D(C_6F_5-F)$ is 6.3 ev (145 kcal), i.e., it is larger by 40 kcal than $D(C_6H_5-H)$ which is equal to 104 kcal. Thus they concluded that C_6F_6 is more stable than benzene.

Steel and Whiffen[2] studied spectroscopically (Raman and infrared) gaseous and liquid perfluoro-benzene, and made the assignment of frequencies with the vibrational modes. Using the results, these authors[3] calculated the force constants and compared them with those of benzene. They found some of the force constants to be the same in perfluoro-benzene and benzene, while some of the perfluoro-benzene force constants are larger than those of the benzene. Thus they also concluded that perfluoro-benzene is more stable than benzene.

It is very unfortunate that we cannot find extensive thermodynamic data (vapor pressure, density, critical point data, specific heat, entropy, etc.) for perfluoro-phenyls. An extensive experimental study should preceed the theoretical studies of perfluoro-phenyls.

At present, we are calculating the thermodynamic properties of gaseous perfluoro-benzene from its partition function. The results will be compared with those of gaseous benzene in order to see which is more stable.

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STATISTICAL MECHANICAL STUDIES ON BIPHENYL

(by Y. L. Wang)

For simplicity, we first study gaseous biphenyl. For the statistical mechanical studies of the substance, it is necessary to know the configuration of the molecule. A precise discussion on the planarity of the molecule is found in the Status Report No. 1 on "Theoretical Studies on Liquid Coolants" by Nakamura, Kwun, Ree, and Eyring,[1] and extensive references on this problem are given therein. It can generally be concluded that the gaseous molecule is not planar; the angle between the two benzene rings if about 45°. The rings perform torsional vibration about this equilibrium position.

Steele and Lippincott[2] determined the fundamental frequencies of the molecule by Raman and infrared spectroscopic studies. These authors found 59 out of the 60 fundamental frequencies. The missing frequency should be of the ring-ring torsional vibration. In constructing the partition function of the molecule, therefore, we approximated the missing frequency by taking it equal to the ring-ring internal rotation. This kind of approximation will introduce a negligible error in the results.

The partition function of biphenyl, f, is represented by

$$f = \left[f_{trans}\, f_{rot}\, f_{int.\ rot}\, f_{vib} \right]^N$$

where, the translational partition function, f_{trans}, indicates the usual

translational partition function after the communal factor (1/N!) was taken

into account. By using this partition function, the thermodynamic quantities,

C_v, H, S, F are calculated, the results agree very well with the available

data in the literature.[3]

The equilibrium constants at various temperatures and pressures for the

system,

$$12 \text{ C (graphite)} + 5 \text{ H}_2 \text{ (gas)} \rightleftarrows \text{C}_{12}\text{H}_{10} \text{ (gas)}$$
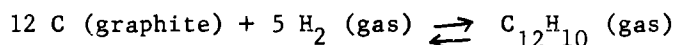
as well as for similar systems for terphenyls are required in nuclear reactor

technology for estimating the safety conditions in the operation. The above

calculated table of the thermodynamic functions has been used to calculate

the temperature variation of the equilibrium constant. Cagle[4] also

calculated the equilibrium constant at various temperatures from experimental

thermodynamical data. The results obtained by these two methods agree very

well.

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MOBILE BOND ORDERS OF m-TERPHENYL

(by S. Kwun)

In a previous report,[1] we reported the mobile-bond order of biphenyl

and p-terphenyl calculated by the Huckel approximation as well as by the

SCF-LCAO-MO theory. The mobile-bond-order calculations of m-terphenyl

are performed following exactly the same procedure used in the previous

report. That is, we solve the 18x18 secular equation for the pi-electron

system of m-terphenyl; this secular equation is decomposed into a 10x10 and

an 8x8 secular equation by using group theory. At present, all the elements

of the secular equation have been obtained by using the m-terphenyl data

calculated by the Huckel theory and reported in the previous report.[1] We

are going to solve the secular equations with use of the computer in the

Campus, the mobile bond orders will then be calculated.

<div align="center">REFERENCE</div>

1. T. Nakamura, S. Kwun, T. Ree, and H. Eyring, "Theoretical Studies on

    Liquid Coolants," AEC (Arco) Project, AT(10-1)-1076, Status Report No. 1,

    University of Utah, 1962.

<div align="center">RADIATION CHEMISTRY OF POLYPHENYLS</div>

<div align="center">(by M. K. Lo)</div>

The stabilities of biphenyl and terphenyls for thermal, beta, gamma and

neutron radiations are very interesting in physical chemistry. The bond-order

calculations conducted in this Laboratory have supplied part of the fundamental

knowledge necessary for studying the radiation chemistry of polyphenyls.

Especially, the changes of bond properties in excited states are very necessary

for the interpretation of radiation chemistry. The calculations for the

excited states have been started by the theoretical group.

Lo has recently joined the group, and is doing the literature survey on

the behavior of polyphenyls under various types of radiation.

<div align="center">STUDIES OF THE SYSTEMS Fe-O-BIPHENYL AND Fe (III) CHLORIDE-BIPHENYL</div>

<div align="center">(by William F. Cagle)</div>

The problem of inorganic iron-oxygen and iron carbon species found in the

reaction have received attention. The compound Fe_3O_4 is formed directly from

Fe_2O_3 and biphenyl. The other products are CO_2 and H_2O with small (1 - 2 %

<div align="center">-116-</div>

at 500°C) but increasing amounts of CO as the reaction temperature is increased. A study of the reaction of Fe_3O_4 to form iron carbides with CO is now underway. This formation proceeds rapidly in the presence of relatively small amounts of hydrogen.

Our bibliography of the reactions of biphenyl and the polyphenyls is now complete and final proofreading is underway. Distribution of this will be made in the near future.

Our study of the iron (III) chloride-biphenyl system has now progressed to the decomposition of the yellow complex compound previously described. A mixture of organic compounds results from the decomposition and it is now in the process of separation. A full report on these materials will be made in the near future.

We now have a program for the "best fit" of $\triangle F$ as a function of the Kelvin temperature and will be able to evaluate the probable error in the data used to calculate $\triangle F_T$ for the reaction $12C_{(s)} + 5H_{2\ (g)} = C_{12}H_{10(g)}$ as well as other data used in our calculations.

ORGANIC DECOMPOSITION STUDIES AND ANALYTICAL

CHEMICAL DEVELOPMENT AT AECL - STATUS REPORT

R. F. S. Robertson
Research & Development Branch
Whiteshell Division
Atomic Energy of Canada Limited

This report summarizes the work done at AECL in the above fields since the June 1961 Conference at Ispra.

COOLANT DECOMPOSITION

Present Status

Irradiations with 1 Mev Electrons - W. D. Mackintosh

Irradiations of Santowax OM (1.7 % biphenyl, 64.6 % o-, 31.5 % m-, 2.2 % p- terphenyl) have continued. Three series of irradiations were carried out: -

(a)   A series of irradiations at 375°C with total dose varied from 4.4 to 105 w - hr/g.

(b)   As above at 450°C and doses from 4.48 to 26.4 w - hr/g.

(c)   Irradiations at constant dose of 8.8 w - hr/g but temperature varied from 350°C to 450°C.

The results may be summarized: -

(i)   Stability as measured by HB content agreed well with that found for Santowax R at Harwell. Results for total gas and H_2 are higher.

(ii)   The ratio of o- to m- terphenyl drops with dose from 2.05 initially to 1.15 for 105 w - hr/g.

(iii)   At 450°C the rate of HB formation is considerably faster than at 375°C.

(iv)   At 8.8 w - hr/g the amount of HB formed increased slightly with increasing temperature from 350°C to 405°C. Above 405°C there is a marked increase in HB formation.

Data substantiating the above conclusions will be presented at the conference.

Reactor Irradiations - A. W. Boyd

Attention during the past year has been focussed on obtaining a proper understanding of reactor dosimetry and not many irradiations have

been performed beyond those outlined in CI-220. As a result of the calorimetric measurements of dose rates proper G values will be applied to all previous in-reactor irradiations. Comparative results will be unchanged and the following conclusions can be drawn: -

(i)   As with electron irradiations, above a certain temperature the HB production rate increases sharply for a given dose. This temperature is about 380° C for a mixture of fast neutrons and γ-rays as compared to 405° C for electrons.

(ii)   In irradiation of OM mixtures the ratio of o- to m- terphenyl decreases with increasing dose.

1. 1. 3   Reactor Dosimetry - A. W. Boyd

The energy deposition due to reactor radiation in various positions in NRX is being studied by a number of methods.

The principal method is calorimetry and calorimeters of three different designs have been constructed and tested. The calorimeters being used at present, which appear to be quite satisfactory, consist simply of a sample with attached thermocouples suspended inside an aluminum can to which thermo-couples are also attached. In operation the calorimeter is positioned in the reactor and the sample and can temperatures are recorded continuously. The energy deposition in the sample is calculated from the specific heat of the sample material and the rate of rise of the sample temperature at the point when the sample and can temperatures are the same. Calorimeters with six different samples have been constructed; aluminum, magnesium, graphite, polyethylene, polystyrene and beryllium.

Some of the values obtained with these calorimeters will be given on the data sheets. The precision with metal samples is of the order of one percent, but the precision is less - about five percent - with the plastic samples, due to thermal gradients in the latter.

The second method of dosimetry was the use of a graphite walled ion chamber with carbon dioxide gas at atmospheric pressure. The ion current measured agreed within ten percent with the value calculated from graphite calorimeter measurements.

The third method of dosimetry was the measurement of hydrogen production in cyclohexane. Irradiations (at Brookhaven) of cyclohexane using cyclotron helium ions and deuterons indicate there is only a small decrease in the hydrogen yield as the track density increases. Irradiations of cyclohexane with fission fragments (at Chalk River) gave a $G(H_2) = 7.5$ - about 50% higher than the value of 5.5 for electrons or gamma rays. Ir-radiations of cyclohexane in various reactor positions in which the fast neutron components differ, ranged from ∿ 3.5 to 4.5. The fast neutrons produce fast protons which are intermediate in track density between the

1.1.3 (Cont'd)

cyclotron and fission fragment irradiations. All these results are somewhat contradictory and further work is needed. It does not appear however that cyclohexane is a suitable compound for accurate reactor dosimetry.

1.2 Future Program

1.2.1 Terphenyl Irradiations

We feel that the work on electron and in-reactor irradiation of terphenyls and Santowax OM is nearing completion. Some tidying up and confirmation of doubtful results will be done, including the following:-

(a)   Fission fragment irradiation of terphenyls in NRX.

(b)   Further examination of fast neutron effects as a function of temperature in a region of NRX where at least 50% of the energy deposition is from fast neutron interactions.

(c)   The radiolysis of triphenylene with 1 Mev neutrons.

(d)   Differential yields of HB and gases under electron irradiation at 375°C for terphenyls which contain deliberately added HB. The HB will be produced in NRX.

(e)   Measurement of the average molecular weight of HB factions previously produced in electron and NRX irradiations.

1.2.2 Pyrolysis

A detailed study of the pyrolysis of pure terphenyls and of those containing HB will be started. Particular attention will be paid to the formation rate of HB and the possible effects of impurities.

1.2.3 Partially Hydrogenated Santowax OM

Some thought is being given to the use of partially hydrogenated Santowax OM, which is liquid at room temperature, for a start-up coolant for WR 1. Radiolysis and pyrolysis as a function of temperature will be studied together with the effect of the decomposition product on viscosity, vapour pressure etc.

2. ANALYTICAL METHODS

2.1 Present Methods

A brief review of present methods is given. More details about newer techniques will be presented in data sheets.

(a)   Gas Chromatography

Techniques followed are outlined in AECL Report CI-219 "The Routine Determination of Biphenyl, o-, m- and p- Terphenyl

in OCDRE Type Coolants by Gas Chromatography" by W. D. Mackintosh, O. A. Miller and L. J. Cornett.

(b)   High Boilers

The micro sublimation method used at Chalk River is based on the one developed at Harwell. It consists of heating 50 mg of the irradiated Santowax OM sample in an evacuated tube at a controlled temperature (175°C) for a fixed time (4 hours). This method gives good reproducibility and has been tested using synthetic mixtures of OMRE high boilers and Santowax OM. In addition to the high boiler analyses, the biphenyl and terphenyl content of all irradiated samples are determined by gas chromatography and it is planned to carry out molecular weight determinations on all sublimates and residues.

(c)   Molecular Weight

A Mechrolab osmometer has recently been purchased and is now being used.

(d)   Vapour Pressure

Our new apparatus consists of a cell in which the sample can be heated to a desired temperature, joined by a small bore tube to a pressure transducer which feeds a signal to a continuous strip recorder. Temperatures in the cell are obtained by means of thermocouples. A 40 cc sample is used. Temperature control is to better than $\pm 1°C$ and the reproducibility of the apparatus is ± 1 psi over the entire range. Within this precision the vapour pressure of ortho- and meta- terphenyl was found to agree with the published Monsanto data. The results for biphenyl above 51 psia also agree although at lower pressures the agreement was not as good. No inter-laboratory comparisons have been made using materials from the same sources.

(e)   Water

Water is determined by Karl-Fischer titration following separation from the organic coolant by azeotropic distillation with dried chloroform. The method will detect as little as 10 ppm water with a reproducibility of ± 5 ppm. At present a reliable water-in-coolant standard is being sought. The entire procedure is characterized by careful attention to the exclusion of atmospheric moisture.

(f)   Iron

The use of a proportional counter has improved the X-ray fluorescence technique to 0.5 ± 0.5 ppm. This method now supersedes dry ashing with calorimetric finish which was much slower and less accurate. There is evidence that some Fe was lost during the ashing process.

(Cont'd)

A note describing this method has been accepted for publication by "Applied Spectroscopy".
Iron in fouling films is determined by an nitric-perchloric ashing followed by an o- phenanthroline colorimetric procedure. Minor metallic constituents are determined on an emission spectrograph.

(g) Oxygen

Dissolved O_2 is determined along with H_2, N_2, CH_4 and CO by gas chromatographic analysis of gas that has been vacuum-stripped from coolant samples. The method can detect 0.05 cc/kg O_2 in organic coolant.
No other types of oxygen analyses have been done so far, but fast neutrons activation techniques will soon be in use to determine total O_2 in the coolant. The lower limit of detection is expected to be 100 ppm. Helium-3 activation analysis as described in Anal. Chem. 34, 329 (1962) may also be investigated. Ultimate sensitivity is reported to be less than one part per billion.
Functional group analysis for ketones, phenols and other oxygen-bearing substances is beginning immediately.

(h) Chlorine

The chloride content of organic coolants is determined by activation analysis. The sensitivity of the method is in the ppm range but is dependent upon the manganese content of the coolant, since manganese activity can mask the desired activity.

(i) Air Pollution Analysis

The concentration of terpnenyls in NRX Reactor air is being determined by infra-red spectrometry and gas chromatography.

(j) Carbon: Hydrogen in Fouling Films

A standard Pregl combustion apparatus has been used to analyze fouling film samples with a standard deviation of ± 0.8% C and ± 0.4% H.

Future Plans

Emphasis will be placed on determination of chemically bound O_2.

In addition we will investigate the possibility of installing a gas chromatograph on a loop as an in-line analytical instrument to monitor O_2 H_2, N_2, CH_4 and CO concentrations on a continuous or semi-continuous basis.

Improvements in all analytical techniques will be sought, especially those for water and iron content of the coolant.

	Prof.	Tech.
Coolant Decomposition	2	2
Analytical	2	3-1/2

There is also a group of 3 professionals and 2 technicians pursuing fundamental radiation chemical investigations in organic liquids and vapours.

VAN DE GRAAFF IRRADIATIONS I MeV ELECTRONS
RADIOLYTIC DECOMPOSITION
375°C 707°F

SANTOWAX R. AERE 350°C 662°F

G (-COOLANT)

% HB
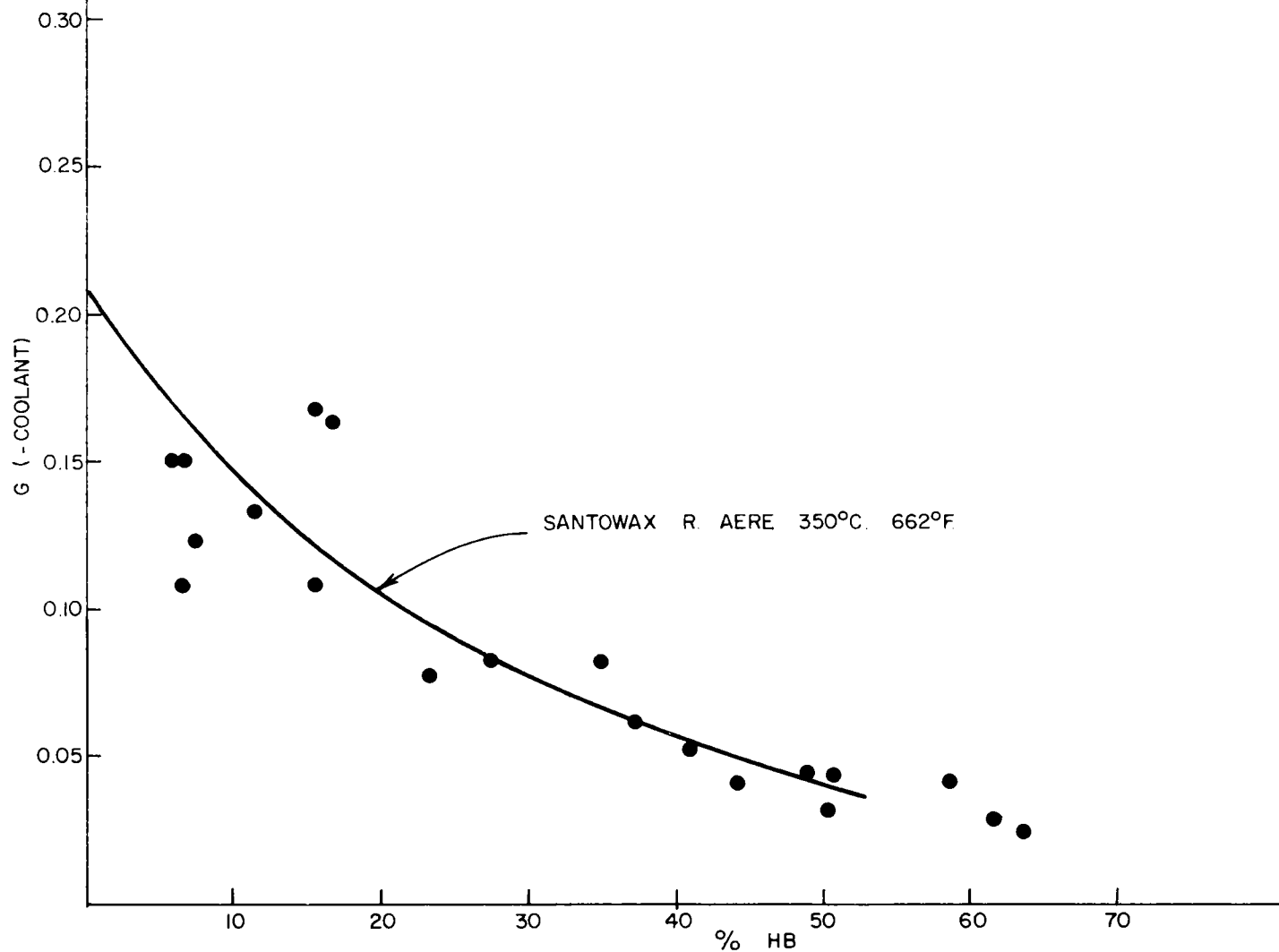
VAN DE GRAAFF IRRADIATIONS I MeV ELECTRONS
TEMP. 375°C AND 450°C

% RESIDUE (100 - Σ Φ_2 Φ_3's)
BY GAS CHROMATOGRAPHY

375 °C

375 °C

450 °C

% HB MICRO - SUBLIMATION

PERCENT HB AND PERCENT 100 - Σ

DOSE WATT - HR /g
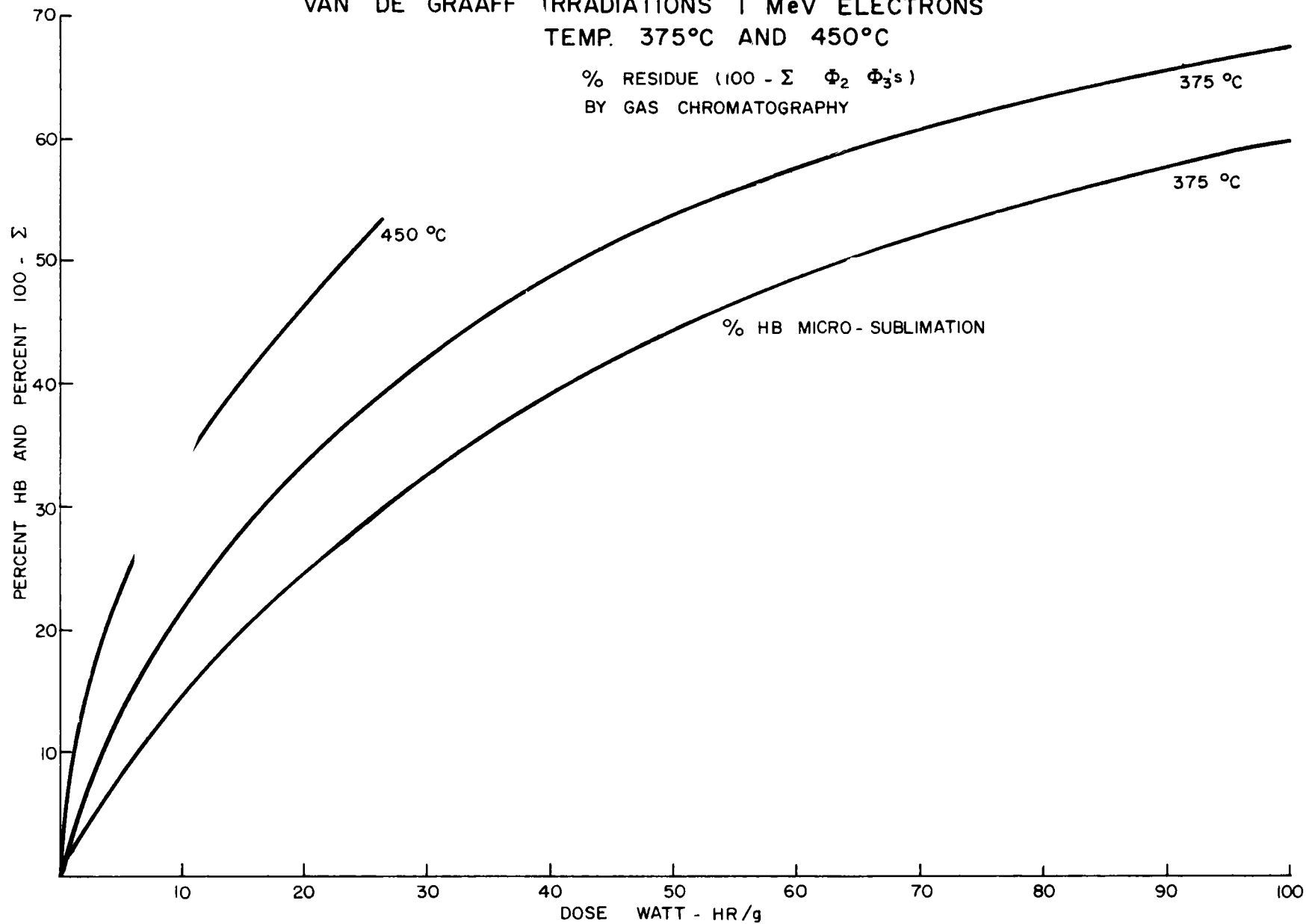
I MeV ELECTRON IRRADIATION OF SANTOWAX O.M.
8.8 WATT HRS/g.
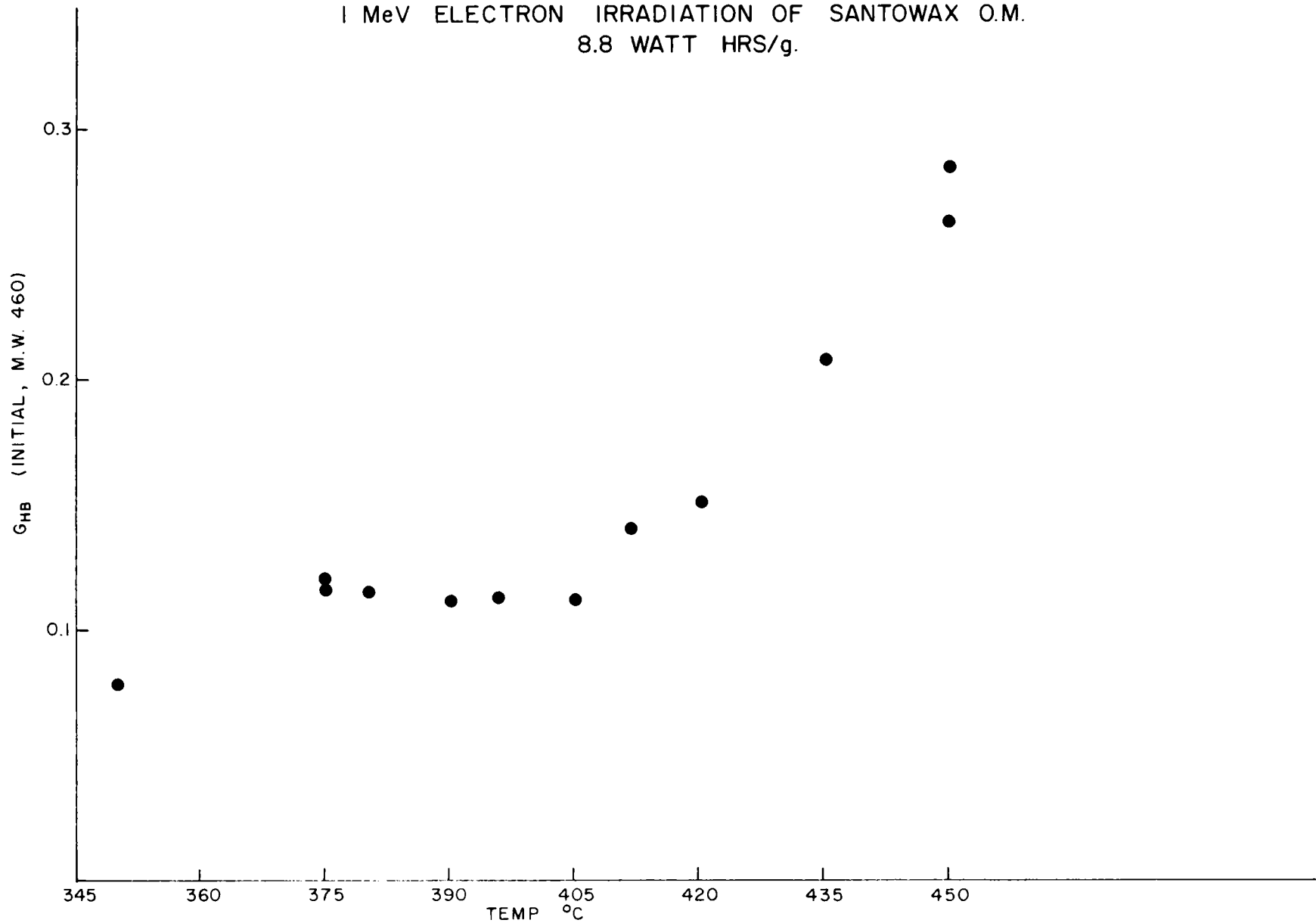
VAPOUR PRESSURE APPARATUS
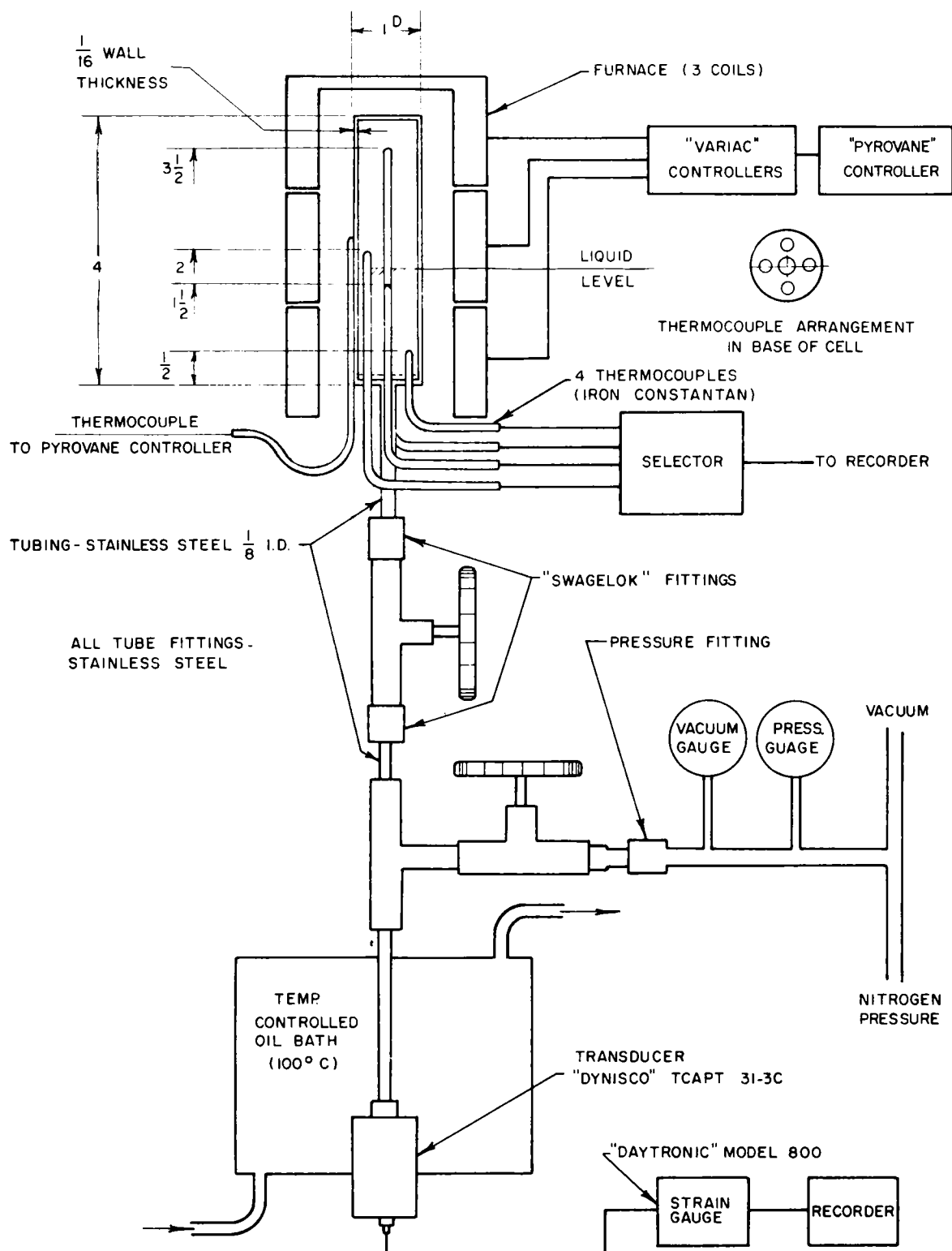
THE DETERMINATION OF IRON IN TERPHENYL

BY X-RAY EMISSION SPECTROSCOPY

A note describing this method has been accepted for publication by "Applied Spectroscopy". A modification of the original technique provides a rapid, simple method for the determination of iron in terphenyl with a limit of detection of 0.5 μg Fe/g of sample where the limit is defined as the value of iron concentration giving a count rate equivalent to three times the standard deviation of the background count.

Homogeneous standards were prepared by dissolving the cupferron complex of iron in terphenyl. Samples as received for analysis usually contain tars and are viscous liquids at room temperature. Samples are melted at 95°C, mixed thoroughly and poured into plastic sample cups which are then capped with 1/4 mil mylar and inverted on a flat surface cooled by liquid nitrogen. To prevent segregation, the samples are counted on the X-ray spectrograph before they thaw.

The instrument used is a Norelco Universal Vacuum X-ray spectrograph with a tungsten target X-ray tube, LiF analysing crystal and flow proportional counter. Following are the conditions used: 2θ angle, line (Fe Kα) 57.50°, bkgd. 60.00°; counter HV, 1475v; P10 Gas flow - 1.5 ft^3/hr; Power 50 KV, 50 ma; pulse height analyser, base line 10v, window width 9v; total count not less than 16,000 for the line and 8.000 for the background.

The standard deviation of the method at the 99% confidence level is 0.5 μg/g of sample or 2% of the iron concentration, whichever is greater. The precision for samples containing particulate iron in concentrations greater than about ten μg/g is somewhat decreased because of the difficulty in obtaining homogeneous samples.

THE DETERMINATION OF WATER IN TERPHENYL COOLANT

BY KARL FISCHER TITRATION

A 25g. sample is contacted with 50 ml. of dried (1) chloroform and the water extracted is separated by azeotropic distillation (2). About 50 ml. of chloroform-sample solution is distilled at a rate of about 1 ml./min.

The separated water is automatically titrated after mixing with 50 ml. of pre-titrated methyl alcohol in the conventional Karl Fischer manner using a Beckman KF-3 "Aquameter". Precision of the method is $\pm$ 5 ppm. Absolute accuracy is not proven because of lack of success in preparing suitable terphenyl water standards.

The entire procedure is characterized by careful attention to the exclusion of atmospheric moisture (3).

Notes:

(1)   The reagent grade chloroform is double-distilled and the still-bottoms are passed through two molecular sieve drying columns. Distillation alone results in a titration blank larger than the titration due to the moisture in the sample, when the moisture level is 20 ppm or less. The molecular sieve columns reduce the blank to about half the titration due to the moisture in the sample.

(2)   It is important that no dead volumes exist in the azeotropic distillation apparatus. The three necessary standard-taper joints should be fitted with teflon liners.

(3)   Samples are obtained from the experimental loops via hypodermic needles attached to sampling by-passes, onto which a dried glass sample bottle capped with a 1/8" - thick neoprene septum is pushed. The remainder of the operation is done in a large dry-box with a controlled relative humidity of 10%. All samples go directly to the dry box. Most water determinations are made within a few hours of sampling, and on no account is the elapsed time longer than three days.

COMPARISON OF IRON FOUND IN COOLANT BY

THREE DIFFERENT METHODS OF ANALYSIS

No.	Sample	Dry Ashing Colorimetric (ppm Fe)	Wet Ashing Colorimetric (ppm Fe)	X-Ray Spect -Direct (ppm Fe)	Ratio (Dry ash/X-ray)
1	AC-10-8	0.9	1.9	2	0.44
2	E-20	18.3	23.0	18	1.02
3	CA1	63.3	69.5	70	0.90
4	CA41	2.2		3	0.74
5	CA42	1.7		2	0.86
6	CA47	5.8		3.5	1.7
7	CA52	3.5		3.5	1..0
8	#15	35.3		29	1.2
9	X-7-18-4	0.57		1.5	0.38
10	CA59	4.2		5.0	0.84
11	CB59	6.2		4.5	1.4
12	CA60	2.5		3.5	0.84
13	CB60	5.6		5.5	1.01
14	CA61	2.5		3.0	0.84
15	CB61	4.5		6.0	0.75
16	CA62	1.8		3.5	0.52
17	CB62	3.0		4.5	0.67
18	CA63	1.8		3.0	0.60
19	CB63	1.8		3.5	0.51
20	CA64	4.3		4.5	0.96
21	CB64	7.2		14	0.51
22	CA69	2.4		4.0	0.59
23	CB69	4.2		6.0	0.70
24	CA70	5.3		6.5	0.82
25	CB70	4.7		6.0	0.78
26	CA72	1.8		2.5	0.70
27	CB72	2.5		4.5	0.55
28	CA73	1.7		1.5	1.11
29	CB73	1.8		2.0	0.92

Results of Calorimetric Measurements
of Energy Deposition Rates in the NRX Reactor.

Annulus in Graphite Reflector
Thermal neutron flux 7.5×10^{12} n cm^{-2}. sec^{-1}

Material	Energy Deposition mw g^{-1}	
	1960	1962
Graphite	15	
Terphenyl	21	
Polyethylene	27	
Biphenyl d_{10}	19	
Aluminum	24	23
Magnesium		17

Empty Lattice Position in Core
Thermal neutron flux 4.0×10^{13} n cm^{-2} sec

Material	Energy Deposition mw g^{-1}
Polyethylene	350
Aluminum	300
Magnesium	290

Centre of Annular Uranium Fuel Element
Thermal neutron flux 5.0×10^{12} n cm^{-2} sec^{-1}

Material	Energy Deposition mw g^{-1}
Graphite	69
Polyethylene	188
Polystyrene	139
Aluminum	60
Magnesium	60
Beryllium	73

FOULING ON FUEL SURFACES

W. M. Campbell

All the fouling studies are carried out at Chalk River in the X-7 loop in NRX and out-reactor in one large and three small loops. Nine runs have been made in the large (250-0-1) out-reactor loop. The results are condensed in Figure 1. A number of conclusions can be drawn from these results, as follows:

1. With a constant inlet temperature the fouling rate increases with the hot surface temperature.

2. Clay purification reduces the fouling.

3. Reducing the water content from 300 to 40 ppm decreases fouling.

4. Some impurity (chlorine suspected) greatly increases fouling by promoting the mass transfer of iron.

5. Most of the fouling films are high in iron - over 75% iron and iron oxide with some as high as 99%.

The fouling films formed under irradiation in the X-7 loop are very similar to those formed out of the reactor. Thus the work at present is mainly in the out-reactor facilities but further in-reactor work will be needed once a clearer picture of the fouling mechanism is obtained.

Many metallographic cross sections of the films have been taken. The results to date show that in "clean" systems mass transfer of iron from the cool walls to the hot surface is the most important contributor to the fouling film. This is proved by mass balances on the iron in the deposit and in the coolant. The organic portion of the deposit is thought to be due to some coolant decomposition product which is unstable above 375°C. It should be emphasized, however, that all the above results apply to "clean" coolant made up from OMRE high boilers and Santowax OM. It contained <5 ppm iron and <8 ppm ash and probably a very low particulate content.

There is strong evidence that iron is being carried atom-by-atom rather than as particulates. Iron mass transfer therefore must be taking place through some intermediate "carrier" such as an organic molecule which can form a complex with iron that is more stable at the bulk coolant temperatures than at the hot wall temperatures. As yet, the "carrier" has not been identified but we suspect that chlorine may play a prominent part in the transfer.

Since iron mass transfer is so prominent one obvious way to reduce it is to remove the iron by some means such as aluminizing the steel. Some experiments have been done in an autoclave where both aluminum and iron were in contact with the coolant. Iron but no aluminum was found in the deposit indicating that this method of reducing fouling may have promise. More work is planned.

One serious problem in all these studies is that a synthetic coolant is being used. There is an urgent need to repeat this work with coolant which has been irradiated at the correct temperature and in the absence of air to give the correct distribution of decomposition products. This may be particularly important for the organic part of the film mentioned above.

Fouling Program

The following are the main items planned:

1. Further runs in the 250-0-1 loop with fresh coolant to get further information on the unexpected behaviour of Run 9 and to determine what the fouling rate is when the coolant is purified to an acceptable reactor condition.

2. A study of coolant impurities such as water content, chlorine content, etc. in the small loops.

3. A comparison of the crud deposition between an all-steel system using different surface conditions and an all-aluminum system.

4. Small loop runs with irradiated coolant to check the effects of irradiation and coolant decomposition products.

5. The X-7 fuel irradiation will be followed from a fouling standpoint and any pertinent out-reactor findings will be checked out.

6. The more fundamental study of iron-hydrocarbon complexes and their stability under different conditions will continue.

Coolant Cleanup

Several possible methods have been tried but treatment with Attapulgus clay at about 350°C appears best. Experiments are being carried out to establish the best design conditions. After an initial cleanup the bed capacity is greater than 10,000 lbs. coolant per pound of clay. The ash content and PCFT have been maintained at < 2 ppm and 6 mg respectively with this treatment. The relation of the capacity of the clay to the water content of the coolant, the amount of exposed iron in the system, etc. will be studied in the future. The impurity which is removed by the clay is unknown but at present no work is planned to investigate this.

Molecular sieves are used to remove water and they appear to be effective down to at least 30 ppm. In a reactor water would be removed in a degasser so no work is planned to investigate the behaviour of the molecular sieves.
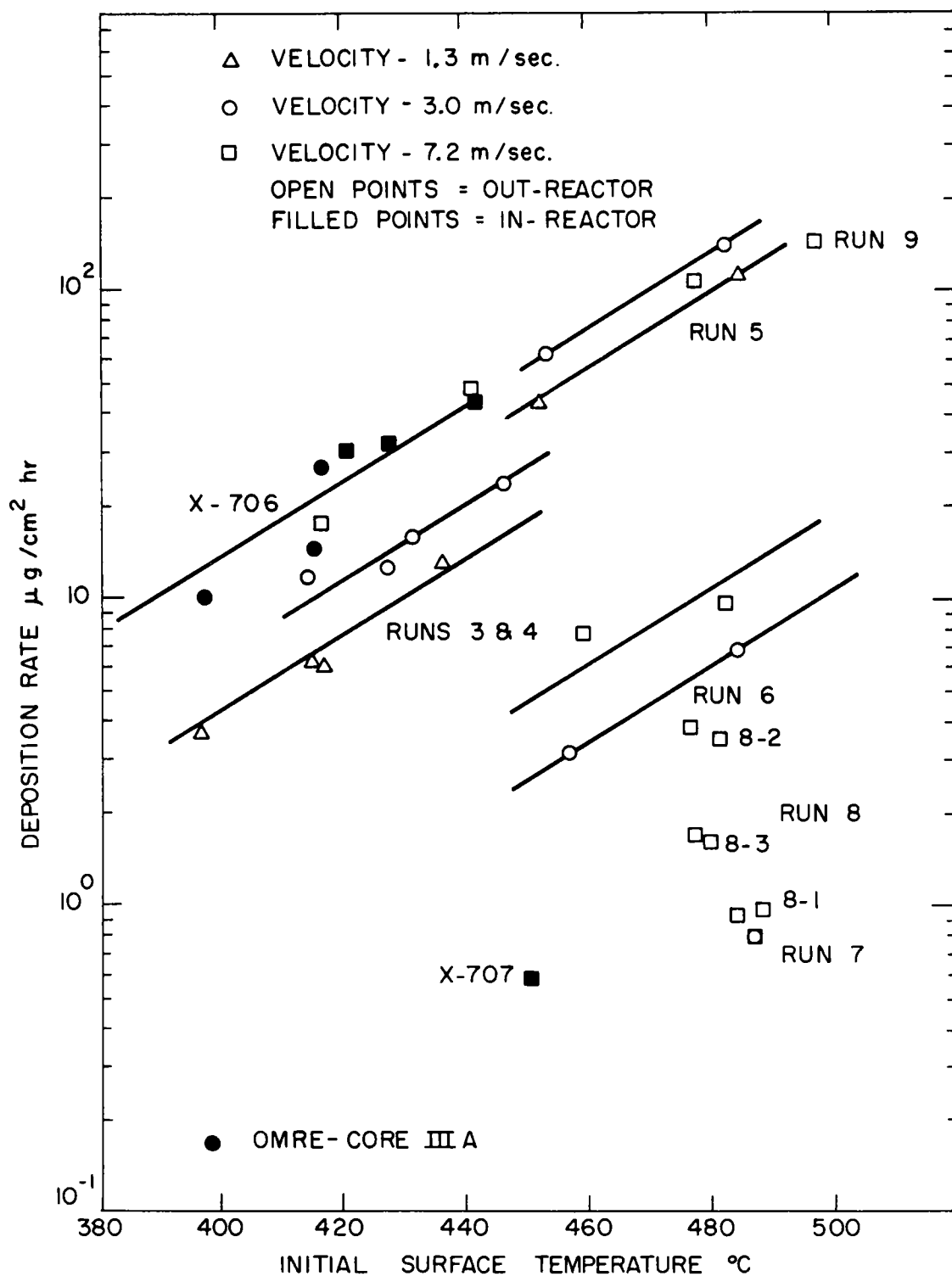
Figure 1

WR-1 STATION DATA

MAY 1962

Labels on the figure:

SHIELDED TRANSFER FLASK

NEW FUEL STORAGE ROOM

TRUCK BAY

FLASK MAINTENANCE & DECONTAMINATION STATION

FUEL WASHDOWN STATION

FUEL BUNDLE & HANGER ROD ASSEMBLIES

SHIELD PLUG

ROTATING DECK PLATES

SPIGOT PLATE

NEW FUEL BUNDLES

REMOVABLE PLUG

SERVICE SPACE

SPENT FUEL STORAGE BAY

OUTLET FEEDERS

INSULATION

FUEL STORAGE TUBE

FUEL TRANSFER MECHANISM

TOP SHUT DOWN SHIELD

WATER TIGHT DOOR

CALANDRIA

STAND-BY STORAGE BLOCK

WORKING BAY

C. CROSS

RADIAL SHIELD

DUMP SPACE

FUEL CAN

BOTTOM SHUT DOWN SHIELD

INSULATION

SERVICE SPACE

INLET FEEDERS

CUT-AWAY OF OTR REACTOR

-138-

SUMMARY OF WR-1 STATION DATA

ISSUE #3   15 MAY 1962

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CANADIAN GENERAL ELECTRIC COMPANY LIMITED

SUMMARY OF WR-1 STATION DATA

000. GENERAL

001	PROJECT		ORGANIC COOLED, DEUTERIUM MODERATED TEST REACTOR.	1
002	OWNER		AECL	1
003	DESIGNERS			
	REACTOR PLANT		CGE	1
	CONVENTIONAL PLANT		SHAWINIGAN ENG.	1
004	SITES & IMPROVEMENTS			
	LOCATION		WHITESHELL NUCLEAR RESEARCH ESTABLISHMENT, MANITOBA, EAST BANK OF WINNIPEG RIVER.	1
	TOTAL SITE AREA		10,000 ACRES	1
	WATER SUPPLY		WINNIPEG RIVER	1
	NEAREST HIGHWAY		HIGHWAY NO.11	1
			WEST SIDE OF RIVER	1
	NEAREST MANITOBA HYDRO POWER		SEVEN SISTERS STATION	1
			7 MILES FROM SITE.	1
	NEAREST RAILROAD		8 MILES	1
	DISTANCE FROM WINNIPEG		52 MILES N.E.	1
	ACCESS ROADS		8 MILES REQUIRED PLUS BRIDGE OVER RIVER.	1
			MANITOBA GOVT. RESPONSIBLE.	1
	PURCHASE OF PROPERTY		AECL	1

100. GENERAL INSTRUMENTATION & CONTROL

101	CONTROL AREA PANEL & EQUIPMENT		
	SUPPLIED BY	CGE	1
	CONTROL PANELS	HORSESHOE ARRANGEMENT.	
		SHEET METAL STRUCTURES.	1
	CONTROL CONSOLE	RECTANGULAR ARRANGEMENT.	1
	RELAY PANELS	SELF SUPPORTING FORMED	
		METAL CABINETS	1
102	COMMUNICATIONS SYSTEM		
	INTERNAL TELEPHONE SYSTEM	AECL OR SHAWINIGAN	1
	PUBLIC ADDRESS SYSTEM	CGE	1
103	ANNUNCIATION SYSTEM	CGE	1
104	FIRE DETECTION & ALARM SYSTEMS	CGE	1
105	FIXED RADIATION CONTAMINATION		
	AND ACCESS CONTROL SYSTEMS.	CGE	1

110. REACTOR PROTECTIVE SYSTEM

TYPE	MODERATOR DUMP	1
REACTIVITY RATE	-30 MK 1.25 SEC. AFTER	
	DUMP VALVES BEGIN OPENING.	1
TOTAL WORTH	SHUTDOWN TO NO	
	MULTIPLICATION FACTOR	1
TRIPPING VARIABLES	TRIPLICATED INSTRUMENTS	1

120. REACTOR REGULATING SYSTEM

120	REACTOR CHARACTERISTICS		
	XENON POISON		
	- EQUILIBRIUM VALUE	30 MK APPROX.	1
	- INITIAL RATE AFTER SHUTDOWN	15 MK/HR. APPROX.	3
	MODERATOR TEMP. COEFFICIENT		
	- EQUIL. FUEL	-.06 MK/DEG.F APPROX.	1
	FUEL TEMP. COEFFICIENT	-.3 MK/MW POWER	
		INCREASE APPROX.	1
	CHANGE OF REACTIVITY		
	WITH MODERATOR LEVEL		
	- @ 8 FT. LEVEL	8 MK/FT.	1
	- @ 4 FT. LEVEL	50 MK/FT.	1

	REACTIVITY CHANGE WITH COMPLETE		
	LOSS OF CORE COOLANT	20 MK APPROX.	1
	MIN. COLD CRITICAL LEVEL	4 FT. APPROX.	1

121 REACTIVITY CONTROL DEVICES

	POISON OVERRIDE METHOD	MODERATOR LEVEL	
		& TEMPERATURE	1
	POISON OVERRIDE TIME	24 TO 45 MINUTES	1
	AUTO. REACTIVITY CONTROL METHOD	MODERATOR LEVEL	1
	RANGE FOR		1
	- FUEL CYCLE - 8 MK	6 FT. 8 IN. TO 7 FT. 4 IN.	
	- POISON OVERRIDE - 6 MK	7 FT. 4 IN. TO 8 FT. 0 IN.	
	MANUAL REACTIVITY CONTROL METHOD	MODERATOR TEMPERATURE	1
	MAX. POSITIVE REACTIVITY RATE		
	AT CRITICAL	.33 MK/SEC.	1

123 COOLANT TEMP. CONTROL DEVICE

		CONTROL VALVE WITH	
		BYPASS IN MAIN HEX	
		PROCESS WATER LINE	1

124 REACTOR POWER REGULATING SYSTEM

	MODES OF CONTROL	LOW MODERATOR LEVEL	1
		LOW POWER	1
		ON POWER	1
	CONTROLLED VARIABLES	MOD. LEVEL, PERIOD, LOG	
		& THERMAL POWER	1
	ACCURACY OF CONTROL	3 % ON THERMAL POWER	
		CONTROL	1
	TYPE OF SYSTEM	SINGLE CHANNEL & STANDBY	
		POWER CUTBACK ON HIGH	
		INDIVIDUAL FUEL ROD	
		OUTLET TEMPERATURE.	1

125 COOLANT TEMP. REGULATING SYSTEM

	CONTROLLED VARIABLES	REACTOR COOLANT INLET	
		TEMPERATURE.	1
	ACCURACY OF CONTROL	5 DEG.F	3
	TYPE OF SYSTEM	SINGLE CHANNEL IN EACH	
		COOLANT SYSTEM	

220. COMMON PROCESSES & SERVICE SYSTEMS

221   PROCESS WATER & INACTIVE DRAINAGE

221.01 GENERAL
        PROCESS & DOMESTIC WATER UP TO A TERMINAL POINT OUTSIDE THE
        BUILDING,& INACTIVE DRAINAGE EXTERIOR TO THE BUILDING, WILL
        BE SUPPLIED BY SHAWINIGAN.   1

221.02 PROCESS WATER
        DISTRIBUTION WITHIN BUILDING   CGE   1
        TEMPERATURE   74 DEG.F MAX.   1
        TREATMENT   STRAINED & INTERMITTENTLY
                                                CHLORINATED   1
        QUANTITY & PRESSURE   8300 IGPM @ 50 PSIG   1

221.03 INACTIVE DRAINAGE WITHIN BUILDING   CGE   1

221.04 DOMESTIC WATER & FIXTURES
        WITHIN BUILDING   SHAWINIGAN   1

222   ACTIVE DRAINAGE

222.01 ACTIVE DRAINAGE OUTSIDE BUILDING   SHAWINIGAN   1

222.02 ACTIVE DRAINAGE WITHIN BUILDING   CGE   1

223   FIRE WATER SYSTEM

223.01 SHAWINIGAN WILL SUPPLY THE FIRE WATER SYSTEM UP TO A
        TERMINAL POINT OUTSIDE OF THE BUILDING   1

223.02 DISTRIBUTION WITHIN BUILDING   CGE   1
        TEMPERATURE   74 DEG.F MAX.   1
        TREATMENT   STRAINED   1
        TOTAL QUANTITY & PRESSURE   2600 IGPM @ 100 PSIG   1
        QUANTITY AVAILABLE FOR EMERGENCY   1400 IGPM   1
        STANDBY SERVICE   1
        QUANTITY AVAILABLE FOR FIRE PROT.   1200 IGPM   1

224   VENTILATION

224.01 GENERAL
        CGE WILL SUPPLY NORMAL & ACTIVE VENTILATING SYSTEMS
        WITHIN THE BUILDING INCLUDING THE STACK CONNECTION
        EXCEPT OFFICE AREA VENTILATION   1

224.02	MAIN BUILDING SUPPLY		
	SOURCE	OUTSIDE AIR	1
	QUANTITY	38,370 CFM	1
	TEMPERATURE	70 DEG.F	1
224.03	REACTOR AREA DISTRIBUTION		
	TOTAL FLOW	20,920 CFM	1
224.04	MAIN BUILDING EXHAUST		
	TOTAL EXHAUST RATE	43,555 CFM	1
	TYPE OF FILTRATION	ABSOLUTE FILTERS	1
224.05	ORGANIC SYSTEMS ROOM VENTILATION		
	HEAT REMOVAL LOAD	500,000 BTU/HR	1
	INLET AIR TEMP.	60-75 DEG.F	1
	OUTLET AIR TEMP.	90-105 DEG.F	1
	FAN CAPACITY	15,000 CFM	1
224.06	CONTROL ROOM AIR CONDITIONING	SHAWINIGAN	1
224.07	OFFICE SYSTEM VENTILATION	SHAWINIGAN	1
225	COMPRESSED AIR - INSTRUMENT		
225.01	SHAWINIGAN WILL SUPPLY INSTRUMENT AIR UP TO A		
	TERMINAL POINT OUTSIDE THE BUILDING.		1
225.02	INSTRUMENT AIR WITHIN BUILDING		
	DISTRIBUTED BY	CGE	1
	INSTRUMENT AIR PRESSURE	100 PSIG	1
	INSTRUMENT AIR QUANTITY	200 CFM	1
	AIR MASKS PRESSURE	25 PSIG	1
226	COMPRESSED AIR - PROCESS		
226.01	SHAWINIGAN WILL SUPPLY PROCESS AIR UP TO A		
	TERMINAL POINT OUTSIDE THE BUILDING.		
226.02	PROCESS AIR WITHIN BUILDING		
	DISTRIBUTED BY	CGE	1
	PROCESS AIR PRESSURE	100 PSIG	1
	PROCESS AIR QUANTITY	200 CFM	1
228	DEMINERALIZED WATER		
	SHAWINIGAN WILL SUPPLY THE DEMINERALIZED WATER SYSTEM UP		
	TO A TERMINAL POINT OUTSIDE OF THE BUILDING.		1
	DISTRIBUTION WITHIN BUILDING	CGE	1
	SOURCE	DOMESTIC WATER SUPPLY	
	QUANTITY	15 IGPM	1

229   MATERIALS HANDLING

229.01   GENERAL
SHAWINIGAN WILL SUPPLY ALL CRANES & FIXED LIFTING DEVICES
REQUIRED FOR NORMAL OPERATION OF OTR. AECL WILL SUPPLY ALL
LIFTING FACILITIES REQUIRED OUTSIDE BUILDING, INCLUDING
FLOATS & TRUCKS. CGE WILL SPECIFY CRANE CAPACITIES.

229.02   REACTOR HALL CRANE
CAPACITY	45 TON	1
SPAN	75 FT.	
AUXILIARY CAPACITY	10 TON	1

229.03   MELT STATION CRANE
TYPE	MONORAIL	1
CAPACITY	112 TON	1

229.04   PURIFICATION & RELIEF ROOM CRANES.
1 - SPAN	15 FT.	1
CAPACITY	5 TON	1
2 - TYPE	MONORAIL	1
CAPACITY	1 TON	1

229.05   NEW FUEL ROOM CRANE
NEW FUEL ROOM CRANE		2
SPAN	19 FT.	1
CAPACITY	1 TON	1

300.00   REACTOR GENERAL

*NOTE   1
SINCE WR-1 IS AN EXPERIMENTAL REACTOR,SUCH MAJOR FEATURES
AS TYPE OF FUEL,COOLANT MATERIAL,& COOLANT TUBE DESIGN
WILL BE CHANGED FROM TIME TO TIME. CONSEQUENTLY THE REACTOR
COMPONENTS & SYSTEMS HAVE BEEN DESIGNED TO ACCOMMODATE A
TOTAL REACTOR THERMAL POWER OF 40 MW EVEN THOUGH THE FIRST
CORE WILL ONLY PRODUCE 30 MW NOMINAL. IN THE FOLLOWING
TABULATION,THEREFORE,MOST OF THE ITEMS APPLY TO THE 30 MW
FIRST CORE DESIGN. THOSE ITEMS WHICH APPLY SPECIFICALLY TO
THE 40 MW MAX. DESIGN ARE MARKED WITH AN ASTERISK.

PROVISION HAS ALSO BEEN MADE FOR A FUTURE 60 MW CORE. A
POSSIBLE CONFIGURATION WOULD BE A 55 CHANNEL CORE WITH 3
COOLANT CIRCUITS.

300.01	REACTOR TYPE	URANIUM OXIDE FUEL	1
		HEAVY WATER MODERATOR	1
		ORGANIC COOLANT	1
		VERTICAL CONSTRUCTION	1
300.02	FIRST CORE		
	TOTAL NUCLEAR THERMAL POWER	30.0 MW	1
	REACTOR THERMAL POWER	28.6 MW	1
	MODERATOR HEATING BY RADIATION	1.4 MW	1
	TOTAL MODERATOR HEATING	1.6 MW	1
	NO. OF FUEL SITES	37 SITES @ 2.23 ENRICH.	1
	POWER/SITE — MAX. FUEL RATING	1.07 MW	1
300.03	CORE DATA		
	LATTICE TYPE	TRIANGULAR	1
	LATTICE PITCH	9.25 IN.	1
	CELL AREA	478 SQ.CM.	1
	NOMINAL CORE RADIUS	29.54 IN. - HOT	1
	NOMINAL CORE HEIGHT	7 FT.	1
	ACTUAL CORE HEIGHT	8 FT.	1
	MAX. THERMAL NEUTRON FLUX	1.08 BY 10 TO THE 14	1
	AVE./MAX. AXIAL FLUX	.637	1
	AVE./MAX. RADIAL FLUX	.776	1
300.04	REFLECTOR DATA		
	MATERIAL	CALANDRIA HEAVY WATER	1
	FORM	LIQUID ANNULUS	
	NOMINAL RADIAL THICKNESS		
	@ CENTER	23.46 IN. - HOT	1
	AXIAL THICKNESS	NIL	1

300. FUEL CHANNELS GENERAL

301   FUEL CHANNEL EQUIPMENT

301.01   FUEL CHANNEL ASSEMBLY

GEOMETRY

OVERALL LENGTH	33 FT. 1.5 IN.	1
MAX. O.D.	6.5 IN.	1
APPROX. WT.- DRY WITHOUT FUEL	400 LB.	1
- WITH FUEL & COOLANT	700 LB.	1
AXIAL GROWTH 70 - 800 DEG.F	2.02 IN.	1

STRESS TUBE IN CORE SECTION @ 70 DEG.F

I.D. MIN. - MAX.	3.250 - 3.260 IN.	2
WALL MIN. - MAX.	.0315 - .0385 IN.	1
OVALITY	.010 IN. MAX.	1
STRAIGHTNESS	.050 IN. MAX. T.I.R.	1
NOM. MEAN HOT I.D. @ 666 DEG.F	3.274 IN.	2

FUEL CLOSURE	STANDARD MECHANICAL TYPE CLOSURE.	1
FUEL SUPPORT	FUEL IS HUNG FROM THE FUEL CLOSURE.	1

HYDRAULICS

SYSTEM DESIGN PRESSURE	300 PSIG	1
PRESSURE DROP ACROSS THE SCATTER PLUG REFERENCE COOLANT @ 800 DEG.F & 300 USGPM		
- TOP PLUG	5.2 PSI	1
- BOTTOM PLUG	17.2 PSI	1

THERMODYNAMICS

SYSTEM DESIGN TEMPERATURE	700 DEG.F	2
HEAT LOSS FROM THE PRIMARY COOLANT TO-MODERATOR CIRCUIT.	5.4 KW/SITE	1
- TOP SHIELD.	4.0 KW/SITE	1
- BOTTOM SHIELD	4.7 KW/SITE	3

MATERIAL - END SECTIONS	CARBON STEEL	1
- CORE SECTIONS	ASTM A269-60-TP347 STAINLESS STEEL.	2

301.02   SHIELDING PLUGS

GEOMETRY
OVERALL LENGTH
    - TCP PLUG   5 FT. 3-5/8 IN.   1
    - BOTTOM PLUG - UPPER SECTION   8 FT. 11-3/4 IN.   1
                 - LOWER SECTION   2 FT. 5-1/8 IN.   1
MAX. O.D.
    - TOP PLUG   7-3/4 IN. - TOP FLANGE   3
    - BOTTOM PLUG - UPPER SECTION   3-3/4 IN.   1
                 - LOWER SECTION   7-1/4 IN. - BOT. FLANGE   3
APPROX. WT. - IN AIR
    - TOP PLUG   300 LB.   1
    - BOTTOM PLUG - UPPER SECTION   100 LB.   1
                 - LOWER SECTION   100 LB.   1

HYDRAULICS
COOLANT   D20   1
APPROX.COOLANT FLOW RATE
TOP & BOTTOM PLUG   300 LB./HR./SITE   1
PRESSURE DROP ACROSS THE PLUG @
300 LB./HR. - TOP & BOTTOM PLUG   10 PSI/PLUG   1

THERMODYNAMICS - APPROX. HEAT
GENERATION - TOP & BOTTOM PLUG   3 KW/SITE/PLUG   1

MATERIAL - CONTAINERS   304L STAINLESS STEEL   1
         - SHIELDING   CARBON STEEL & MASONITE   1
                                                         50-50 RATIO   1

310. REACTOR STRUCTURES GENERAL

312.01	CALANDRIA	2
FORM	VERTICAL CYLINDER TAPERED	1
	ON TOP SIDE WITH DISHED	1
	TOP & BOTTOM.	1
MATERIAL	304L STAINLESS STEEL	1
NOMINAL INSIDE DIAMETER	8 FT. 10 IN.	1
OVERALL HEIGHT	16.5 FT.	1
WALL THICKNESS	1/2 IN.	1
TOP HEAD THICKNESS	3/4 IN.	1
BOTTOM HEAD THICKNESS	1-3/8 IN.	1
DRY WEIGHT - LESS EXTENSIONS	32,200 LB.	3
DESIGN INTERNAL PRESSURE FOR		
EMERGENCY CONDITIONS.	100 PSIG	1
DESIGN EXTERNAL PRESSURE.	15 PSIG	1
SUPPORTS	6 ROD HANGERS AROUND	
	CIRCUMFERENCE ATTACHED	
	TO RADIAL SHIELD ASSEMBLY	1
DUMP PORTS	PERIPHERAL,	1
	13 SQ.FT. AREA.	1
DUMP SPACE VOLUME	70% OF CORE D2O	1
MODERATOR HEIGHT	7 FT. PLUS 1 FT. FOR	3
	CONTROL.	1
SPRAY COOLING	D2O COOLING FOR WALLS,	1
	TOP HEAD & COOLANT TUBES.	
312.02	CALANDRIA TUBES	2
MATERIAL	ASTM B210-60T-ALLOY 5052	2
	ALUMINUM.	2
QUANTITY	13	2
- 4.755 IN. I.D.		2
- .060 IN. WALL		2
QUANTITY	24	2
-3.695 IN. I.D.		2
- .050 IN. WALL		2
315.01	SHIELDS	
TYPE OF CONSTRUCTION	ALTERNATE LAYERS OF	
	STEEL & LIGHT WATER	1
MATERIAL	CARBON STEEL	1
TOTAL THICKNESS - TOP	4 FT.	1
- BOTTOM	4 FT.	1
- RADIAL	1 FT.	1
APPROX. DRY WEIGHT - TOP	160,000 LB.	1
- BOTTOM	112,000 LB.	1
- RADIAL	145,000 LB.	1

315.02   DECK PLATE
           FORM   3 PIECE CONSTRUCTION
                                                WITH CONCENTRIC AND
                                                ECCENTRIC ROTATING
                                                SEGMENTS.   1
           OPERATION   MANUAL   1
           THICKNESS   2 FT.   1
           APPROX. SEGMENT WEIGHTS
              - STATIONARY RING   118,000 LB.   1
              - CONCENTRIC PLATE   52,000 LB.   1
              - ECCENTRIC PLATE   36,000 LB.   1
           MATERIAL   CARBON STEEL   1

320. HEAVY WATER & HELIUM PROCESS SYSTEMS

321	MODERATOR SYSTEM		
321.01	FLOW THRU CALANDRIA	530 IGPM	3
	REACTIVITY DESIGN D20 TEMP. IN CALANDRIA.	130 DEG.F	3
	TEMP.RANGE THRU WHICH MODERATOR IN CALANDRIA WILL BE VARIED.	90-200 DEG.F	1
321.02	MODERATOR PUMPS	2	1
	TYPE	CENTRIFUGAL,MECHANICAL SHAFT SEAL.	1
			1
	CAPACITY/PUMP	50% OF FULL FLOW	1
	MOTOR RATING	50 HP	1
321.03	MODERATOR HEAT EXCHANGER	1	1
	TYPE	U TUBE	1
	TUBE MATERIAL	INCONEL	1
	MAX. HEAT DUTY	12.0 MILLION BTU/HR.	1
	SURFACE AREA	1000 SQ.FT.	3
	FLOW RATE	530 IGPM	3
	NOMINAL TEMP. IN & OUT	120 & 95.6 DEG.F	3
321.04	DUMP TANK	CYLINDRICAL	1
	DIAMETER	7 FT.	1
	VOLUME	500 CU.FT.	1
	MATERIAL	304 S.S. CLAD	1
	INLET NOZZLE	10 IN., SCHEDULE 40	1
	OUTLET NOZZLE	10 IN., SCHEDULE 40	1
322	AUXILIARY HEAVY WATER SYSTEMS		
322.01	DEMINERALIZER SYSTEM		
	NO. OF COLUMNS	1	1
	RESIN COLUMN SIZE & TYPE	2 CU.FT. AMBERLITE XE150	1
	ION EXCHANGE COLUMN LIFE	12 MONTHS	1
	FLOW RATE	15 IGPM	1
	CONTROLLED PH	6-7	1
	ION EXCHANGE OPERATION	INTERMITTENT	1
	CLEAN UP TIME CONSTANT & HALF TIME	0.276/HR. & 2.5 HOURS	1
	NO. & TYPE OF FILTERS	1-CLEANABLE EDGE TYPE	1
	FILTRATION FLOW RATE	15 IGPM	1
322.02	HEAVY WATER LEAKAGE COLLECTION		
	NO. OF TANKS	2	1
	CAPACITY/TANK	50 IMP. GALLONS	1
	PUMP CAPACITY	10 IGPM	1

323   COVER GAS SYSTEM

COVER GAS	HELIUM	1
GAS FLOW RATE	90 CFM	1
NO. OF BLOWERS	2	1
BLOWER TYPE	D2O LIQUID RING SEAL	1
MOTOR RATING	10 HP	1
DIFFERENTIAL PRESSURE MAINTAINED		
BETWEEN CALANDRIA & DUMP SPACE.	4.4 PSI	1
TYPE OF DUMP & CONTROL VALVES	BUTTERFLY	1
DUMP VALVE OPENING TIME	0.5 SECONDS	1

380.   HEAVY WATER INVENTORY @ 120 DEG.F

CALANDRIA	31,500 LB.	1
MODERATOR SYSTEM PIPES	2470 LB.	1
MAIN PUMPS	120 LB.	1
DEMINERALIZER CIRCUIT	323 LB.	1
HEAT EXCHANGER	1050 LB.	1
HELIUM PUMPS	106 LB.	1
INSTRUMENTATION SYSTEM	20 LB.	1
APPROX. 3% RESERVE	1111 LB.	1
TOTAL	36,700 LB.	1

330. ORGANIC & GASES PROCESS SYSTEMS

331   REACTOR HEAT TRANSPORT SYSTEM

333.01   THE FOLLOWING ITEMS APPLY TO FUEL FOR THE 30 MW REACTOR   2
         TOTAL FLOW   3.3 MILLION LB./HR.   1
         OUTLET TEMPERATURE   700 DEG.F   1
         INLET TEMPERATURE   640 DEG.F   1
         VELOCITY THRU COOLANT TUBES   30 FT./SEC. MAX.   1
         DESIGN PRESS. DROP ACROSS FUEL   150 PSI   1
         CHANNEL OUTLET PRESSURE   165 PSIA   1

333.32   THE FOLLOWING ITEMS APPLY TO FUEL FOR THE 40 MW REACTOR   2
         *TOTAL FLOW   1
         *INLET TEMPERATURE   1
         *DESIGN PRESS. DROP ACROSS FUEL   1

331.03   HEAT EXCHANGERS   2 - 1 FOR EACH SYSTEM   2
         TYPE   WATER COOLED   1
         *RATING   67.5 MILLION BTU/HR MAX. 1
         TOTAL SURFACE AREA   1000 SQ.FT.   1
         TUBE MATERIAL   CARBON STEEL   1
         COOLANT VELOCITY   1
         COOLANT PRESSURE DROP   20 PSI   1

331.04   PRIMARY COOLANT PUMPS   2 - 1 FOR EACH SYSTEM   2
         TYPE   CENTRIFUGAL , VERTICAL 1
         PUMPING CHARACTERISTICS   4160 IGPM @ 735 FT. HEAD. 2
                                             3120 IGPM @ 840 FT. HEAD. 2
         CAPACITY/PUMP   100% FULL FLOW/LOOP   1
         PUMP MATERIAL   CARBON STEEL   1
         MOTOR RATING FOR EACH PUMP   1000 HP   1

331.05   PRIM. COOLANT DRAIN TANK CAPACITY   2340 IMP. GALLONS   2

331.06   SURGE TANKS   2 - 1 FOR EACH SYSTEM   2
         CAPACITY   200 IMP. GALLONS   1

331.07   FLOODING TANKS   2   2
         CAPACITY   2500 CU.FT. EACH   1
         CIRCULATING PUMP   20 IGPM @ 150 PSI   1
         FLOODING TANK HEATERS   1 EACH @ 15 KW   1
         COVER GAS   NITROGEN   1

332	AUXILIARY ORGANIC SYSTEMS		
332.01	STANDBY COOLERS	2 - 1 FOR EACH SYSTEM	1
	TYPE	WATERCOOLED	1
	RATING	2.0 MILLION BTU/HR MAX.	1
	TOTAL SURFACE AREA	100 SQ.FT.	1
332.02	CHARGING SYSTEM		
	NO. OF PUMPS	1	1
	PUMPING CAPACITY	3 IGPM	1
	MOTOR RATING	1.5 HP	1
332.03	COOLANT PURIFICATION SYSTEM		
	DISTILLATION UNIT	FLASH STILL	1
	OPERATING PRESSURE	0.5 PSIA	1
	PURIFICATION THROUGHPUT	800 LB./HR.	1
	NO. OF PRODUCT TANKS	2	1
	CAPACITY/TANK	250 IMP. GALLONS	1
	KETTLE CAPACITY	210 IMP. GALLONS	1
	FLASH STILL HEATER	33 KW	1
	RESIDUE PUMP CAPACITY	3 IGPM	1
	RESIDUE COOLER	35 SQ.FT.	1
332.04	MAKE-UP MELTING STATIONS	2	1
	TYPE	DRUM HEATERS	1
	HEATER RATING	6 KW	1
	CIRCULATING PUMP CAPACITY	10 IGPM	1
332.05	LEAKAGE COLLECTION SYSTEM		
	COVER GAS	NITROGEN	1
	NO. OF TANKS	2	1
	CAPACITY/TANK	75 IMP. GALLONS	1
	NO. OF PUMPS	2	1
	CAPACITY/PUMP	2 IGPM	1
332.06	DEGASSING & PARTICULATE		
	REMOVAL SYSTEMS	2 - 1 FOR EACH SYSTEM	1
	TYPE OF DEGASSER	OPERATING CONDENSER	1
	DEGASSER VOLUME	350 CU.FT./SYSTEM	1
	MAX. RATE OF GAS REMOVAL	50 STD. CU.FT./SYSTEM	1
	PUMPING CAPACITY	20,000 LB./HR./SYSTEM	1
	NO. OF HOT WELL PUMPS	2/SYSTEM	1
	CAPACITY/PUMP	50% OF FULL FLOW	1
	MOTOR RATING	10 HP	1
	OPERATING PRESSURE	25 PSIA	1
	NO. OF ADSORPTION COLUMNS	2/SYSTEM	1
	TYPE	ATTAPULBUS CLAY	1
	CAPACITY/COLUMN	30 CU.FT.	1
	NO. OF BACK-UP FILTERS	2/SYSTEM	1
	TYPE	SINTERED STAINLESS STEEL	1
332.07	TRACE HEATING	STEAM HOT WATER	1

333   GAS SYSTEMS

333.01   PRESSURE RELIEF SYSTEM
           MAX. CAPACITY
           NORMAL CAPACITY
           NO. OF GAS DELAY TANKS   2
           CAPACITY/TANK   80 CU.FT.
           WATER SCRUBBING CAPACITY

332.02   GAS PRESSURIZING & PURGE SYSTEM
           TYPE   NITROGEN
           SUPPLY   BOTTLES
           NUMBER
           PRESSURE   250 PSIG MAX.

                340. SPECIAL AUXILIARY SYSTEMS

341   THERMAL SHIELD COOLING
           COOLANT   WATER
           HEAT REMOVAL CAPACITY   3.5 MILLION BTU/HR
           MAX. PERMISSIBLE SHIELD TEMP.   140 DEG.F

342   VAULT CONCRETE COOLING SYSTEM
           TYPE   RECIRCULATING WATER
           TOTAL HEAT REMOVAL CAPACITY   150,000 BTU/HR
           MAX. PERMISSIBLE TEMP. IN CONCRETE 130 DEG.F

343   SPENT FUEL BAY CIRCULATION
           CIRCULATION RATE   75 IGPM
           BAY CHANGES/DAY   2
           MAX. WATER TEMPERATURE   125 DEG.F
           TYPE OF PURIFICATION   ION-EXCHANGE & FILTRATION

344   WASH-DOWN SYSTEM
           SOLVENT   XYLENE

360. FUEL HANDLING SYSTEM

350   GENERAL

METHOD OF CHANGING FUEL	AT SHUTDOWN - 6 HOURS	1
FUEL CHANGING DEVICE	VERTICAL TRANSFER FLASK	1
STANDBY STORAGE	26 SITE STORAGE BLOCK	1
FUEL EXAMINATION	EXAMINATION STATION	1
SPENT FUEL STORAGE	UNDER WATER, CAPACITY	1
	OF OVER 110 FUEL RODS.	1
SPENT FUEL SHIPPING	VERTICAL SHIPPING CASK	1

351.01   SHIELDED TRANSFER FLASK

NUMBER OF FLASKS	1	1
FUEL CAPACITY	1 FUEL ROD	1
METHOD OF COOLING	ORGANIC, NAT. CONVECTION	1
TRAVERSE MECHANISM	STATION CRANE	1
MAX. LENGTH	32 FT.	1
MAX. WEIGHT	43 TONS	1

351.02   STANDBY STORAGE BLOCK

CAPACITY	6 WET SITES PLUS 20	
	FUTURE WET OR DRY SITES	
COOLING	ORGANIC, NAT. CONVECTION	1
	TO COLD WATER HEAT SINK	1
OPERATIONAL ACCESS	REACTOR HALL	1

352   FUEL EXAMINATION, HANDLING & STORAGE

NEW FUEL BUNDLE SHIPPING CONTAINER		
WEIGHT FULL	1400 LB.	1
WEIGHT EMPTY	880 LB.	1
CAPACITY	3 BUNDLES	1
NEW FUEL HANGER SHIPPING CONTAINER		
WEIGHT FULL	2000 LB.	1
WEIGHT EMPTY	400 LB.	1
CAPACITY	15 HANGERS	1

352.02   NEW FUEL STORAGE   40 FUEL ASSEMBLIES   1

352.03   FUEL EXAMINATION

METHOD	VERTICAL SCANNING	
	BY PERISCOPE	1
SHIELDING	CLEAR ORGANIC	1
OPERATIONAL ACCESS	REACTOR HALL	1

352.04	FUEL WASH-DOWN STATION		
	CAPACITY	1 FUEL ROD	1
	TYPE	VERTICAL TUBE	1
	METHOD OF WASHING	SOLVENT CIRCULATION	1
	SHIELDING	CONCRETE	1
	COOLING	CIRCULATION	1
	OPERATIONAL ACCESS	REACTOR HALL	1
352.05	SPENT FUEL STATION		
	TYPE	VERTICAL L-SLOT	1
	SHIELDING	CONCRETE	1
	CAPACITY	TO TRANSMIT 1 FUEL ROD	1
	TRANSFER MECHANISM	TROLLEY, CHAIN DRIVE	1
	COOLING	STAGNANT AIR	1
352.06	SPENT FUEL STORAGE	2 WATER STORAGE BAYS	1
		PLUS WATER WORKING BAY	1
	METHOD — OPTIONAL	IN CANS FILLED WITH	
		ORGANIC	1
	STORAGE BAY AREA	240 SQ.FT.	1
	STORAGE BAY DEPTH	18 FT.	1
	WORKING BAY AREA	70 SQ.FT.	1
	WORKING BAY DEPTH	25 FT.	1
	SPENT FUEL STORAGE RACKS	4	1
	CAPACITY — EACH	14 RODS	2
352.07	HANDLING EQUIPMENT	UNDERHANG MANBRIDGE	1
	CAPACITY	1 TON	1
	SEPARATION OF BUNDLES & HANGERS	UNDER WATER	1
	LOADING INTO SHIPPING CASK	UNDER WATER	1
	SPENT FUEL SHIPPING CASK		
	CAPACITY	9 FUEL BUNDLES	1
	WEIGHT	37 TONS	1
	HEIGHT	12 FT.	1
	HANDLING	STATION CRANE	1
353	SERVICING & EMERGENCY EQUIPMENT		
	SHIELDED FLASK SERVICING AREA	FLASK MAINT. PIT	1
	SHIELDED FLASK DECONTAMINATION		
	AREA	FLASK MAINT. PIT	1

370. FUEL

370	**GENERAL**		
	FISSIONABLE MATERIAL	UO2 PELLETS WITH DISHED	1
		ENDS. - 2.23 ENRICH.	1
	FUEL DENSITY @ 20 DEG.C	10.4 TO 10.7 GM./CC.	1
	SHEATH MATERIAL	SAP	1
	TYPE OF ASSEMBLY	19 ELEMENT BUNDLE	1
	BUNDLES/SITE	1	1
	NO. OF BUNDLES IN CORE	37	1
371	**FUEL BUNDLE DATA**		
	FUEL LENGTH	96 IN. - HOT	1
	OVERALL BUNDLE LENGTH	106 IN. APPROX.	1
	COOLANT LINER I.D., COLD - HOT	3.255 - 3.275 IN.	2
	BUNDLE DIAMETER - MAX. - HOT	3.228 IN. - OVER FINS	1
	SHEATH I.D. - MAX., COLD - HOT	.5565 - .5625 IN.	1
	SHEATH O.D. - MAX., COLD - HOT	.6035 - .6095 IN.	1
	SHEATH THICKNESS - AVE.	.0235 IN.	1
	RIB HEIGHT - AVE.	.042 IN. - HOT	1
	RIB WIDTH - AVE.	.043 IN. - HOT	1
	FUEL AREA - AVE., COLD - HOT	.241 - .249 SQ.IN.	1
	SHEATH AREA - AVE., COLD - HOT	.046 - .047 SQ.IN.	1
	ELEMENT TO ELEMENT SPACING		
	- AT ENDS - MAX.	.048 IN.	1
	- AWAY FROM ENDS - MIN.	.038 IN.	1
372	**FUEL TEMPERATURE & RATINGS**		
	MAX. FUEL TEMP. - AT SURFACE	1380 DEG.F	3
	- AT CENTRE	3730 DEG.F	3
	MAX. SHEATH SURFACE TEMP.	895 DEG.F	1
	MAX. - INTERNAL PRESSURE	350-450 PSIA	1
	MAX. SURFACE HEAT FLUX	298,000 BTU/HR.SQ.FT.	3
	LINEAR POWER RATING		
	- OF BUNDLE	7.85 KW/CM. - NORMAL	1
	- OF ELEMENT - Q/4PYE	36.4 W/CM. - NORMAL	1
	HEAT FLUX DISTRIBUTION	SINUSOIDAL IN AXIAL	
		DIRECTION.	
	COOLANT OUTLET TEMP.	700 DEG.F	1
	COOLANT INLET TEMP.	640 DEG.F	3
	AVERAGE COOLANT TEMP.	670 DEG.F	3
	AVERAGE COOLANT VELOCITY	30 FT./SEC.	1
	TOTAL COOLANT AREA - AVERAGED		3
	OVER BUNDLE LENGTH.	2.7950 SQ.IN.	3
	BUNDLE EQUIVALENT DIAMETER	.229 IN.	2
	COOLANT OUTLET PRESSURE	165 PSIA	1
	MAX. PRESSURE DROP AVAILABLE	135 PSI	3

390. COOLANT

390.01 COOLANT GENERAL

MATERIAL	MIXTURE BY WEIGHT OF	1
	70% SANTOWAX OM 70 &	1
	30% RADIOLYTIC TARS	1
DENSITY @ 700 DEG.F	.87 GM/CC	1
SPECIFIC HEAT @ 130 DEG.F	.36	1
SPECIFIC HEAT @ 640 DEG.F	.56	1
FINAL MELTING POINT	152 DEG.F	1
VISCOSITY @ 130 DEG.F	590 C.P.	1
VISCOSITY @ 640 DEG.F	.51 C.P.	1
THERMAL CONDUCTIVITY @ 130 DEG.F	.084 BTU/HR.FT.DEG.F	1
VAPOUR PRESSURE @ 700 DEG.F	39 PSIA	1
MOISTURE CONTENT	LESS THAN 100 P.P.M.	1
ESTIMATED ORGANIC DAMAGE RATE @ 30 MW NOMINAL & 30% HIGH BOILER	11.2 LB./HR.	1

FOR FURTHER PROPERTIES SEE OTR DESIGNERS MANUAL OF   3
ORGANIC COOLANT PROPERTIES, CEI-142, FEBRUARY 9, 1962 BY   3
D.H. CHARLESWORTH A.E.C.L.   3

390.02 COOLANT INVENTORY

REACTOR	3800 LB.	700 DEG.F	1
INLET & OUTLET FEEDERS	8760 LB.	700 DEG.F	1
PRIMARY COOLANT LOOP #1			
HEAT EXCHANGER 2390 LB.			1
SURGE TANK 1360 LB.			1
PUMPS & PIPING 5493 LB.			1
LOOP #1 TOTAL	9243 LB.	700 DEG.F	1
LOOP #2 TOTAL	9243 LB.	700 DEG.F	1
PRIMARY COOLANT SWELL TO DEGASSER #1	5813 LB.	700 DEG.F	3
PRIMARY COOLANT SWELL TO DEGASSER #2	5813 LB.	700 DEG.F	3
DEGASSER CIRCUIT #1 INCLUDING SWELL	11513 LB.	700 DEG.F	1
DEGASSER CIRCUIT #2 INCLUDING SWELL	11513 LB.	700 DEG.F	1
PURIFICATION SYSTEM	7200 LB.	700 DEG.F	3
RESERVE ALLOWED	0 LB.		1
TOTAL TO FILL SYSTEM 30% HB	61,272 LB.	700 DEG.F	3
FLOODING TANKS USING HB-40	322,000 LB.	200 DEG.F	1

500. ELECTRICAL SYSTEMS

GENERAL

SHAWINIGAN ENG. WILL SUPPLY SUB-STATION.   1

510   CABLES, CONDUITS & TRAYS
           TYPE OF CABLE   TW IN SMALL SIZES   1
                                              RW-RH IN LARGE SIZES   1
           MAIN ROUTING   IN CABLE TRAYS   1
           CABLE INTO PROCESS AREAS   VIA CONDUIT   1
           C.S.A. WIRING METHOD CLASS 1 GROUP D DIVISION 2 & CLASS 2   1
           GROUP F DIVISION 2 IN AREAS CONTAINING ORGANIC ABOVE   1
           100 DEG.C, AND IN AREAS WHERE DUST CONCENTRATION MAY GO   1
           ABOVE .035 OZ./CU.FT.   1

520.01   HV BUS & SUPPLIES
           NORMAL SOURCE   WHITESHELL SUB-STATION   1
           STANDBY SOURCE   NONE   1
           VOLTAGE   4160V NOMINAL   1
           FREQUENCY   60 CPS   1
           PHASE   3   1
           LOADS   ABOVE 200 HP   1
           TOTAL LOAD CAPACITY   5000 KVA   1

           DATA ASSUMED FOR 4160V SUPPLY TO OTR.   2
            - MAX. REGULATION   3%, FROM NO LOAD TO
                                              FULL WHITESHELL
                                              TRANSFORMER CAPACITY.   2
            - MIN. SHORT CIRCUIT CAPACITY   45 MVA, CALCULATED USING
                                              TRANSIENT REACTANCES.   2
            - MAX. SHORT CIRCUIT CAPACITY   150 MVA, CALCULATED USING
                                              SUBTRANSIENT REACTANCES.   2

520.02   MV BUS & SUPPLIES
           NORMAL SOURCE   HV BUS VIA 2-1250 KVA
                                              TRANSFORMERS.   1
           STANDBY SOURCE   250 KW DIESEL GENERATOR   1
                                              SUPPLIED BY SHAWINIGAN.   1
           VOLTAGE   600V NOMINAL   1
           FREQUENCY   60 CPS   1
           PHASE   3   1
           LOADS   1/2-200 HP NORMAL   1
                                              1/2-50 HP STANDBY   1

520.03	LV BUS & SUPPLIES		
	NORMAL SOURCE	MV MAIN BUS VIA	
		2- 150 KVA TRANSFORMERS	1
	STANDBY SOURCE	MV STANDBY BUS VIA	
		1-45 KVA TRANSFORMER	1
	VOLTAGE	120V NOMINAL	1
	FREQUENCY	60 CPS	1
	PHASE	1	1
	LOADS	BELOW 1/2 HP	1

520.04	DC BUS & SUPPLIES		
	NORMAL SOURCE	MV BUS VIA RECTIFIER	1
	STANDBY SOURCE	BATTERIES FLOATING ON	1
		BUS, RATED FOR 1 HOUR	1
		OPERATION	1
	VOLTAGE	125V NOMINAL	
	LOADS	INSTRUMENTATION, CONTROL	
		& EMERGENCY LIGHTING.	

520.05   LIGHTING
SHAWINIGAN WILL SUPPLY LIGHTING SYSTEM FROM THE LIGHTING
PANELBOARDS TO & INCLUDING LIGHTING FIXTURES.   1

REACTOR DESCRIPTION
HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

UNITED STATES ATOMIC ENERGY COMMISSION
Savannah River Plant, S. C.

by
Nathaniel Stetson, Director
Civilian Reactor Division
Savannah River Operations Office

The Heavy Water Components Test Reactor (HWCTR) is a heterogeneous
pressure vessel, D_2O-moderated and cooled test reactor constructed
for the AEC's Heavy Water Development Program. The reactor was
designed to provide test facilities for the irradiation of natural
uranium candidate fuel elements being produced in the AEC's program.
The reactor will also test the use of conventional materials and
components in the reactor's cooling system as an approach to reducing
the capital cost of heavy water power reactors.

The reactor is located at a site designated 700-U Area on the Savannah
River Plant. This site was chosen because of the availability of
supporting facilities such as office buildings, roads, water and
utility services. The site also provides access to the reactor on an
unclassified basis within the Savannah River Plant complex.

The HWCTR was built at a cost of $9,750,000, including two isolated
coolant loop systems for liquid and boiling D_2O.

Construction of the HWCTR was begun in April of 1959. A composite
concrete and mild steel containment shell 70' in diameter and 120'
high was constructed in a 60' deep excavation. Concrete was used
below grade. The reactor was installed after the containment building
had been leak tested. Major construction was completed in December
of 1961, and the reactor was made critical on March 3, 1962. Major
delays in construction were encountered as a result of an extensive
steel strike occurring during late 1959 and 1960 and delays in delivery
of the reactor vessel due to fabrication difficulties at the vendor's
shops. At the present, the reactor has completed a series of low
power physics tests and is undergoing modifications and tests in
preparation for power operation.

The HWCTR vessel is fabricated of carbon steel plates clad on the
internal surfaces with Type 304 stainless steel by roll clad tech-
nique. The pressure vessel is designed for 1500 lbs/sq/in pressure

at a maximum temperature of 315°C. It consists of a lower 7' 2" I. D. shell section, an upper 5' 0" I. D. shell section, and a conical transition section, a hemispherical welded on bottom head, and an ellipsoidal bolted on top head. The vessel is 29' 5 1/2" over-all length.

The reactor core contains 12 test fuel assemblies, 24 driver fuel assemblies of enriched uranium, 12 single control rods, one cluster of six control rods, six safety rods, and six instrument thimbles.

The lattice is divided into two regions. Near the periphery of the reactor is a region comprising 24 fuel assemblies of enriched uranium alloyed with zirconium (driver assemblies). The function of the driver assemblies is to provide adequate reactivity for the reactor. These assemblies are arranged in six groups of four assemblies each, all on a circle of 3' 4" diameter. The assemblies are grouped in this way to avoid interference with piping which can be installed for operation of isolated coolant loops at six of the outer test positions. Two of the test fuel positions are currently occupied by bayonets connected to isolated coolant loops.

The driver region surrounds a test region that contains positions for 12 fuel assemblies of natural or slightly enriched uranium (metal or oxide). These positions are arranged on a 7" triangular pitch in two concentric hexagons. The 12 control rods, located in the driver region, are spaced uniformly in a circle of 2' 7" diameter. The six safety rods are spaced uniformly in a circle of 2' diameter between the driver and test regions. The core is bounded in the radial direction by a reflector of D_2O 18" thick and at top and bottom by shield plates. The distance between the shield plates, which defines the core height, is 10'.

Reactor control is provided by mechanical drives attached to boron steel control rods. The drive racks are housed in pressure thimbles which are attached to the top head of the vessel. The racks are attached to extensions of the control rods and are housed in zircaloy guide tubes within the reactor core. The control rod drive mechanism can be delatched from the rods and the head unbolted and removed to provide access to the core for loading.

The reactor is cooled in two circulating loops, each consisting of a 4800 gpm high pressure shaft seal pump and a vertical hairpin type steam generator rated at approximately 30 MW. Coolant enters the vessel through two 16" diameter lines near the top and flows downward through annular channels defined by the fuel and housing tubes of the

reactor fuel elements. The heated coolant immerges at the bottom of the fuel assemblies mixing with the bulk moderator of the reactor core. It is then withdrawn through 10" diameter outlet nozzles at the top of the core region and is circulated by the high pressure pumps through the tubes of the steam generator. Treated light water from wells located in the area is fed into the steam generators. Cooling is accomplished by an open cycle in which the light water steam formed in the generators is vented to the atmosphere. The reactor can produce and dissipate 61 MW of heat at rated conditions.

The reactor has been constructed with two isolated coolant loops installed in test fuel positions. One of these loops will permit operation with liquid D_2O cooling at pressures up to 500 psi above the core pressure. Cooling to this loop is accomplished by a conventional pump and heat exchanger loop.

The other isolated coolant loop permits studies of fuel performance under boiling conditions. An in-line steam quencher is used to condense the outlet steam at qualities ranging from 10 to 30%. Cool D_2O for this quencher is obtained from a conventional cooler operating on a by-pass from the pump discharge. Outlet steam quality can be controlled by variation of the by-pass rate and control of the ΔT across the steam quencher.

Both the main reactor circulating system and the isolated coolant loops are pressurized with helium gas provided by high pressure storage tanks in the area. The boiling isolated loop is pressure equalized to the reactor vessel.

Purification of the heavy water coolant for the reactor is accomplished by a continuous purge. A purge cooler permits the use of low pressure vessels containing filters in mixed bed ion exchange resins. Purification of the isolated coolant loops is accomplished in a common system for both loops by similar means.

The main reactor circulating system including the steam generators is fabricated entirely of carbon steel. Lithium hydroxide is added to maintain pH at greater than 10, and deuterium gas is added to the helium pressurizing gas to react with dissolved oxygen in the system. Both isolated coolant loops are fabricated entirely of stainless steel.

The initial program of operation of the HWCTR will be directed to testing tubular fuel elements of natural uranium metal clad with zircaloy by the co-extrusion process. In addition, uranium oxide tubular fuel elements fabricated by vibratory compaction, swaging, or a combination of the two, will be tested in the HWCTR. One assembly consisting of five CANDU fuel bundles will also be tested in the initial test loading of the HWCTR as part of the Canadian Co-operative Program.

HEAVY WATER COMPONENTES TEST REACTOR (HWCTR)

Purpose:

To permit the full-scale testing, at power reactor conditions, of fuel assemblies and components for a heavy water moderated power reactor fueled with natural uranium. The facility will be able to test 12 fuel elements at simulated power reactor conditions of heat flux, temperature and pressure. The reactor will also provide data on D_2O leakage from shaft seal pumps and conventional components. The primary system is fabricated of carbon steel, and a composite concrete and steel containment structure has been provided to test these methods of reducing reactor costs.

Reactor:

Type	Pressurized Heterogeneous Pressure Vessel, D_2O-Moderated and Cooled, Forced Convection
Power	61 MWt - dissipated to atmosphere as steam
Location	SRP - TC-1 Site (new designation 700-U Area)
Structures	Reactor Building 770-U
	Control Building 774-U
	Health Physics Building 735-U
	Mech. Equipment Building 771-U
	Shops & Offices (existing) 704-U
	Secondary Substation 752-U
	Water Treatment Building 781-U

Reactor Containment Building:

Dimensions	70' diameter x 125' high
	60' below grade
Materials	18" thick post-tensioned concrete (below grade)

Materials (cont'd)	3/4" steel (above grade - vertical sides)
	3/8" steel dome
Design Pressure	24 psig internal, 0.75 psig external
Capacity of storage tank in dome	15,000 gallons
Free volume of building	320,000 cu. ft.
Startup date	March 3, 1962

Reactor Vessel:

Over-all height	29' 5 1/4"
Total weight (empty)	98 tons
Lower shell section	7' 2" I. D.
Upper shell section	5' 0" I. D.
Material	Carbon steel lined with 1/4" 304 stainless steel
Nominal thickness of walls	4 1/2"
Method of pressurization	He blanket gas

Core:

Diameter	3' 4"
Length	9' 5"
Number of fuel positions	36 - 24 driver, 12 test
Driver fuel (tubular)	9% oralloy-zirconium alloy clad with Zr-II or IV
Test fuel	Normal or slightly enriched U metal or UO_2 (tubular or oxide rods)

Lattice:

Pitch	7" triangular
Number control rods	12
Number safety rods	6
Number instrument rods	6
D_2O reflector thickness	18"
Nominal operating pressure	1000 psig
Maximum operating pressure	1500 psig
Maximum coolant temperature	290°C
Maximum specific power (test fuel)	55 MW/T
Estimated fuel lifetime (drivers)	9 months @ approx. 80% innage
Driver fuel loading	25 kgs. oralloy

D_2O Circulating System:

Number loops	2
Piping material	304 stainless steel in reactor shield, carbon steel outside shield

D_2O Inventory (including loops): 63,987 lbs.

Number circulating pumps	2
Capacity, each	4,800 gpm
Drive Motor	250 H. P., 1800 RPM, AC
Backup	15 H. P., DC motor on same shaft

Steam Generators:

Number	2
Type	Vertical U-tube
Material	Carbon steel
Heat transfer surface	2500 sq. ft.
Design pressure (shell)	600 lbs/in.^{2}
Size	4' diameter x 21' long

Isolated Coolant Loops:

Number of possible loop positions	6 (displace test positions)
Number of loops installed	2
D_2O inventory	7,284 lbs.
High pressure D_2O loop	2,250 psia max.
Boiling loop	1,750 psia max.
Outlet steam quality	Variable to 30%

Test Fuel Elements:

Initial charge of test fuel elements	Natural U or UO_2 tubular and rod bundle candidates

LATTICE DIAGRAM

Isolated Coolant
Loop Bayonet Positions

○ Driver fuel assembly (enriched uranium)

◎ Test fuel assembly
(Outer six positions can hold bayonets)

Control Rods

● Black full rod

● Gray full rod

○ Safety rod

● Instrument position

77" diameter inside thermal shield

CONTROL ROD NOZZLE 12

SAFETY ROD NOZZLE 6

TOP GUIDE PLATE

LIQUID LEVEL NOZZLE

NORMAL WATER LEVEL

REACTOR SHELL

STILLING CHAMBER

TOP SHIELD

D_2O INLET

SHIELD SUPPORT RING

SHIELD MUFF

HORIZONTAL THERMAL SHIELD

THERMAL SHIELD

D_2O OUTLET

DRIVER TUBES 24

CONTROL RODS 12

SAFETY RODS 6

INSTRUMENT TUBES 6

CONTROL ROD CLUSTER

MONITOR PINS 36

CONTROL CLUSTER NOZZLE

LOOP CONNECTORS

LOOP PIPING

ISOLATED LOOP BAYONET

LOOP TEST ELEMENT

HWCTR
TEST FUEL ASSEMBLY

THICK WALLED SINGLE TUBE
(0.250")
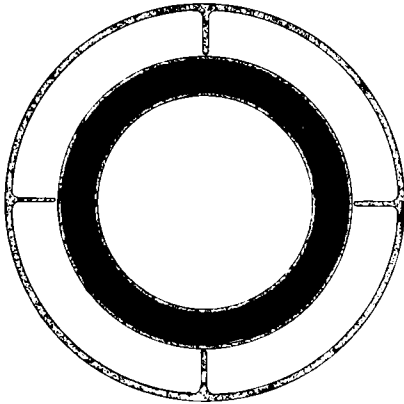
■ U-METAL OR UO₂

▨ ZIRCALOY

THIN WALLED DOUBLE TUBE
(0.130")
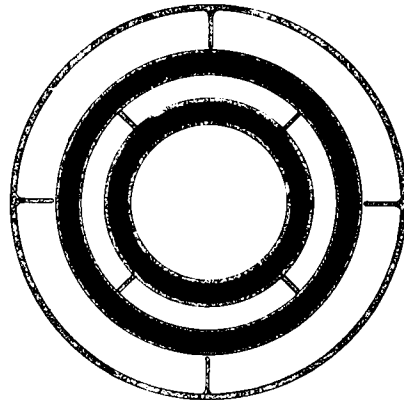
■ U-METAL OR UO₂

▨ ZIRCALOY

19 ROD UO₂ ELEMENT
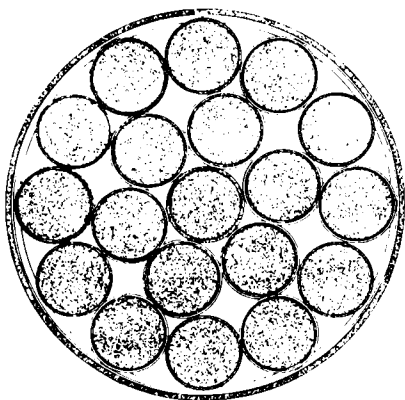
▨ SWAGE COMPACTED UO₂

▨ ZIRCALOY-CLADDING & HOUSING

▢ ZIRCALOY WIRE SPACER WRAP

DRIVER FUEL ELEMENT
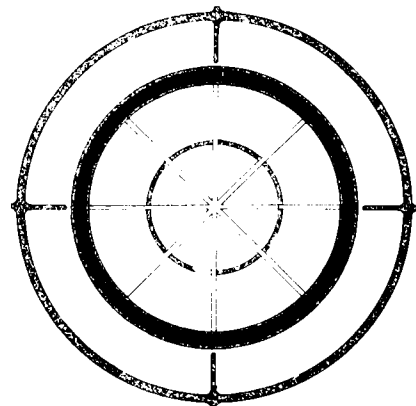
▨ ZIRCALOY-HOUSINGS & CLADDING

■ U-Zr ALLOY- 9.3 W/d ORALLOY-Zr-FUEL

▢ BORON-STAINLESS STEEL -0.6% BO-S.S.
TARGET

HWCTR TEST FUEL IRRADIATION SCHEDULE
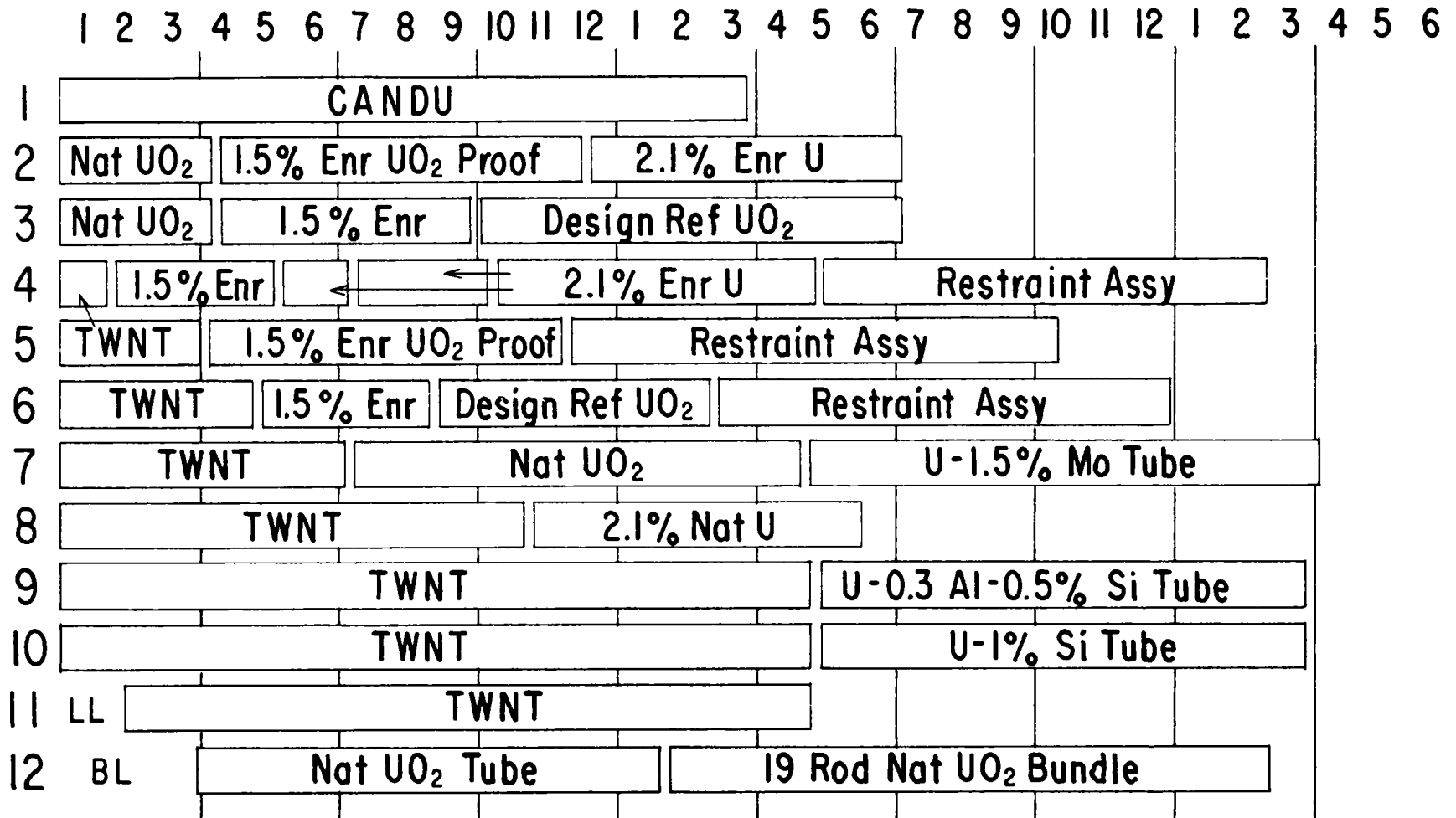
DESIGN AND STATUS

OF THE

PLUTONIUM RECYCLE TEST REACTOR

BY

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HANFORD OPERATIONS OFFICE

INTRODUCTION

The Hanford Plutonium Recycle Test Reactor, the PRTR, is a heavy water cooled and moderated reactor and is designed to demonstrate the technical and economic feasibility of recycling plutonium as a fuel in thermal heterogeneous reactor systems. It is expected that PRTR technology can be utilized in the future in most types of thermal power reactors. Major development problems now being studied fall into four general areas:

1. Economical techniques for fuel element fabrication.

2. Analysis of reactor operation and control.

3. Development of economical fuel separation processes.

4. Economic optimization of fuel cycles.

BRIEF HISTORY

In 1956 the Division of Reactor Development authorized the Hanford Operations Office to initiate a research and development program on the plutonium recycle concept and to support this effort the design of the PRTR was undertaken. In the fall of 1957 Congress appropriated $13,500,000 for complete design and construction of the PRTR. Subsequently $400,000 was appropriated for construction of the Plutonium

Recycle Critical Facility as an integral part of the PRTR building.

This facility will permit physics evaluation of irradiated PRTR fuel

elements and is now being design tested.

Construction of the PRTR was completed on September 12, 1960, and

criticality was achieved on November 21, 1960. The nominal operating

power of 70 MW was reached on July 21, 1961. Accumulated PRTR

exposure to date is approximately 8000 MWD's.

PRTR DESCRIPTION

The PRTR (Figures 1 and 2) is a vertical pressure-tube reactor,

moderated and cooled by heavy water. The maximum thermal power

rating is 75 MW and the normal operating thermal power is 70 MW.

The reactor core is contained in an aluminum calandria 7 feet in

diameter and $9\frac{1}{2}$ feet high. Isolation of the fuel tubes from each

other and from the cold, unpressurized moderator makes possible a

diversified program of fuel irradiation and testing. The heavy water

coolant and moderator provides increased advantages of reactor safety

and permits greater latitude in the selection of fuel configurations

than would be possible with a graphite moderated, water cooled reactor.

CONTAINMENT

The PRTR is housed in a welded steel containment vessel that is 80 feet

in diameter and 120 feet high. The dome is hemispherical and the base

is hemiellipsoidal. Approximately 75 feet of the vessel is above

grade and is designed for an internal pressure of 15 psig. Satis-

factory pressure tests have been made at 18.75 psig, using the
Vallecitos method of leak testing.

The above grade section of the containment vessel is called the "reactor
hall". The reactor is located directly on the center axis of the
reactor hall and below grade elevation. Most supporting equipment is
located in the 180 degree cell below the zero level of the reactor.
The other two quadrants are experimental and instrument cells. Incor-
porated in the shielding wall between the experimental and process cells
is a fuel examination facility (32 x 4 x 6 feet) where irradiated fuel
elements can be examined in air. This unit is not yet in operation.
Normal access to the containment vessel is via a personnel air lock
from the PRTR service building. An emergency personnel air lock and
an equipment access door are also provided. The service building
houses the reactor control room, maintenance shop, mock-up area,
service equipment, fuel loadout facility and offices.

PRIMARY COOLING CIRCUIT (Figure 3)

Heat generated in the fuel elements is dissipated by pressurized heavy
water flowing upward through each of the 85 vertical Zircaloy-2 process
tube assemblies. The process tubes are arranged on an 8-inch equi-
lateral triangle lattice and are connected to ring headers at the top
and bottom faces of the reactor. The process tube operating pressure
is 1050 psig and the system circulation rate is 8400 gpm. The inlet
heavy water temperature is 478° F; the exit temperature is 530° F.
The D_2O coolant is collected in the 10-inch outlet ring header and
passed through the tube side of a horizontal, U-tube, kettle-type,
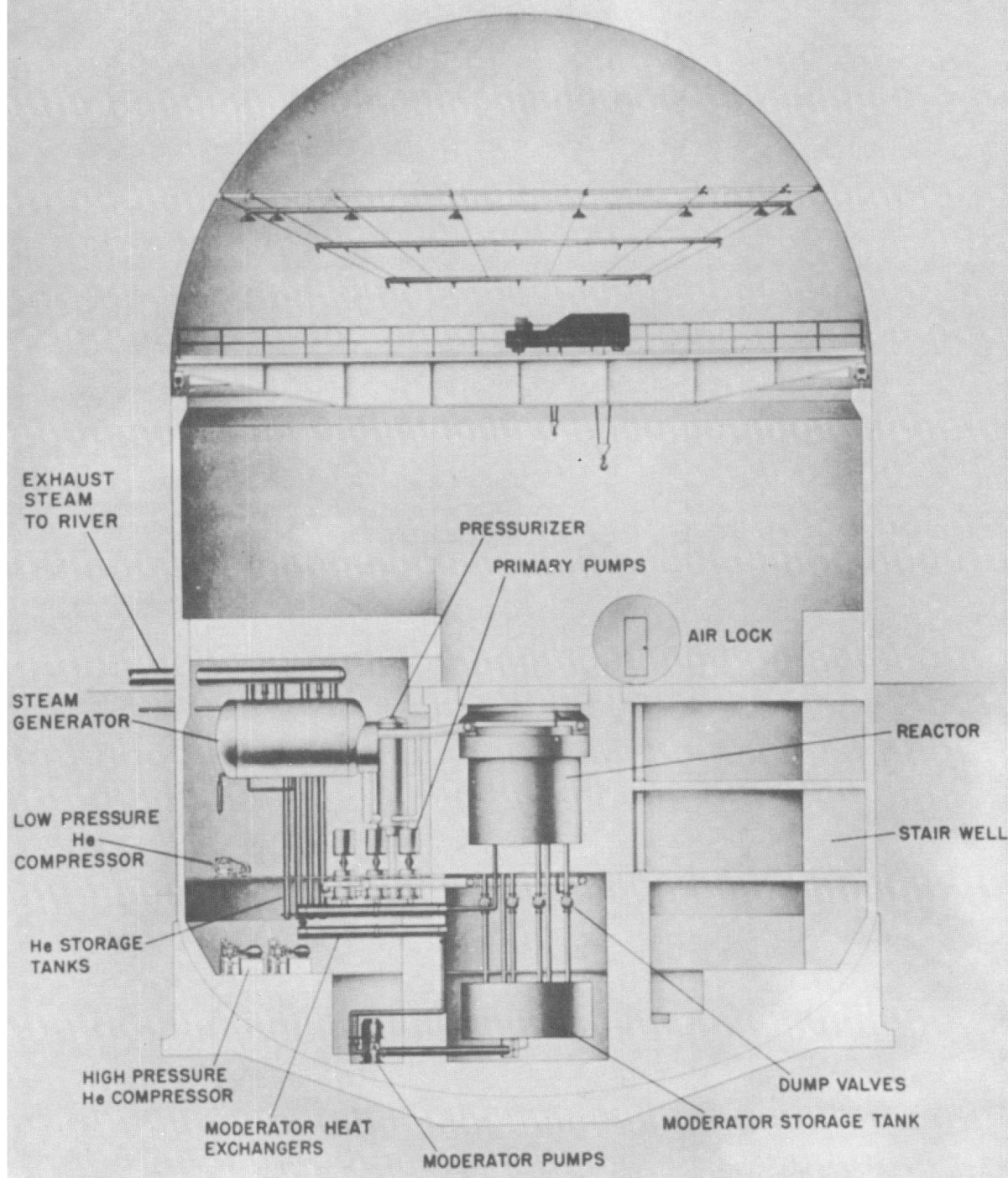
PLUTONIUM RECYCLE TEST REACTOR
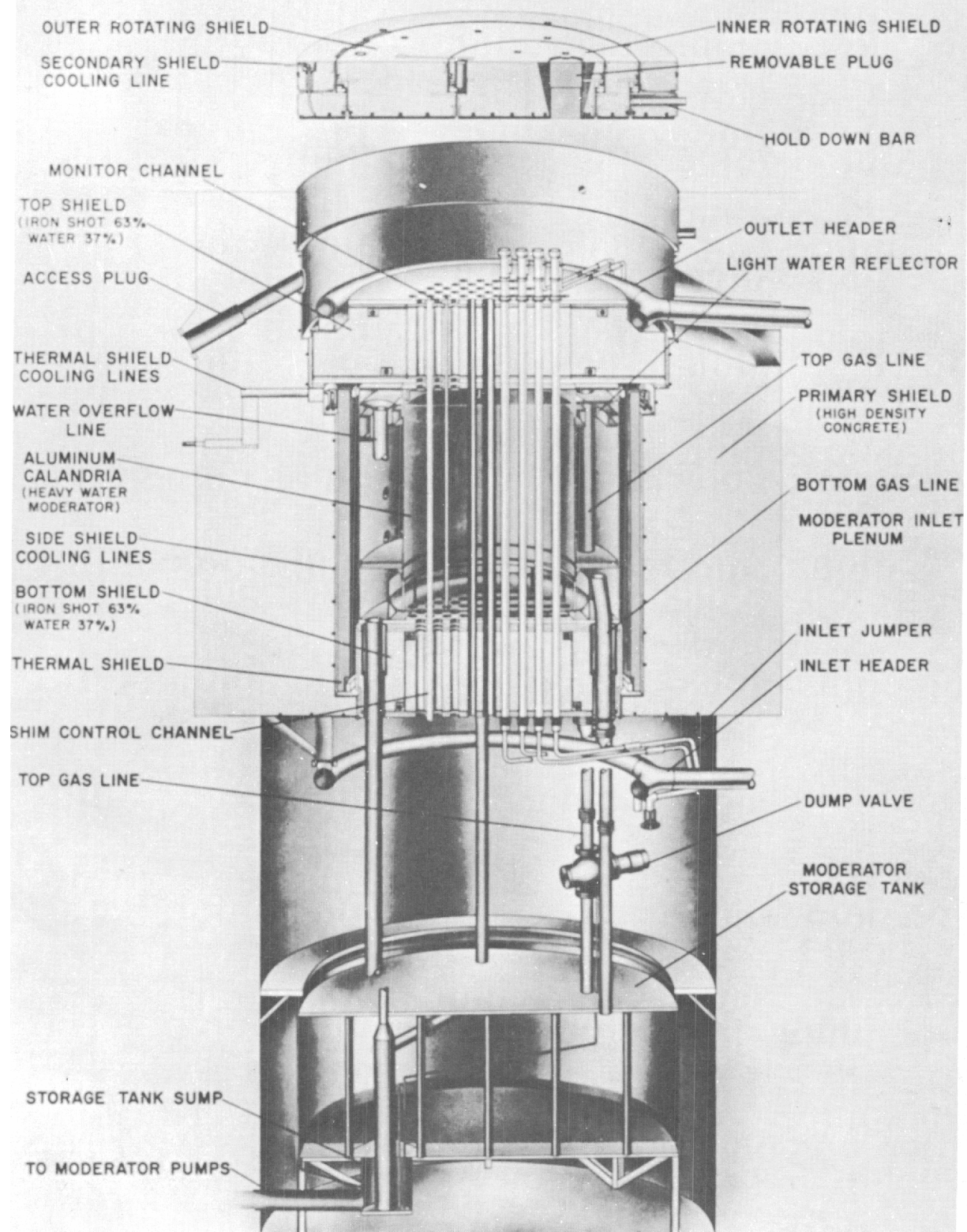

FIGURE 1

PRT REACTOR CORE & SHIELDING

OUTER ROTATING SHIELD

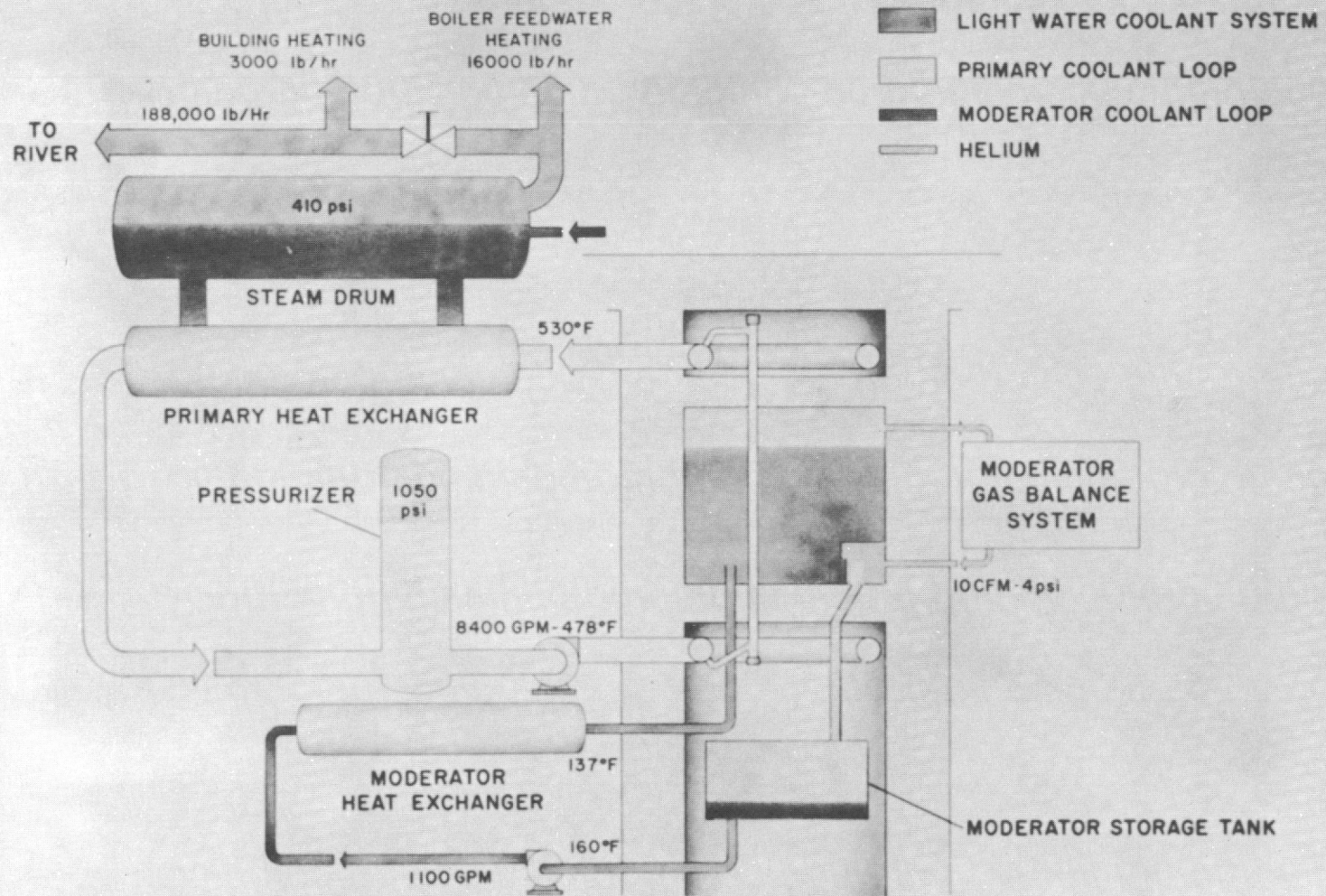SECONDARY SHIELD COOLING LINE

INNER ROTATING SHIELD

REMOVABLE PLUG

HOLD DOWN BAR

MONITOR CHANNEL

TOP SHIELD (IRON SHOT 63% WATER 37%)

ACCESS PLUG

THERMAL SHIELD COOLING LINES

WATER OVERFLOW LINE

ALUMINUM CALANDRIA (HEAVY WATER MODERATOR)

SIDE SHIELD COOLING LINES

BOTTOM SHIELD (IRON SHOT 63% WATER 37%)

THERMAL SHIELD

SHIM CONTROL CHANNEL

TOP GAS LINE

OUTLET HEADER

LIGHT WATER REFLECTOR

TOP GAS LINE

PRIMARY SHIELD (HIGH DENSITY CONCRETE)

BOTTOM GAS LINE

MODERATOR INLET PLENUM

INLET JUMPER

INLET HEADER

DUMP VALVE

MODERATOR STORAGE TANK

STORAGE TANK SUMP

TO MODERATOR PUMPS

FIGURE 2

-178-

PRT REACTOR SCHEMATIC FLOW DIAGRAM

FIGURE 3

combination heat exchanger and steam drum. Single pass cooling by light water on the shell side of the steam generator produces approximately 200,000 lbs. per hour of 425 psig steam. The steam produced is, for the most part, condensed and wasted to the Columbia River. After leaving the steam generator the D_2O coolant passes to the pressurizer where a system pressure of 1050 psig is established by bleeding in high pressure helium. Two 350 horsepower primary pumps (a third is available as a spare) circulate the D_2O back to the process tubes via the inlet distribution ring header. The mechanical seal vertical centrifugal primary pumps are rated at 4200 gpm at 290 ft. TDH and 1750 rpm. Each pump is equipped with a flywheel designed to provide necessary pump decay in the event of electric power failure.

FUEL

The initial PRTR fuel element assemblies consisted of two types of clusters of 19 fuel rods. Externally, all the assemblies were very similar in the sense that each rod consisted of a core about 0.5 in. in diameter and were clad in 30 mil Zircaloy. Thirty PRTR fuel assemblies contained an alloy of plutonium and aluminum, 1.8% by weight plutonium. The remaining 55 fuel assemblies consisted of swaged rods of natural uranium dioxide. Any particular assembly consisted entirely of either Pu-Al or UO_2 rods - there was no mixing. Since startup this pattern has changed from time to time as new experimental elements have replaced portions of the original loading. The plutonium-aluminum alloy and UO_2 fuel assemblies are interchangeable in the reactor within the limitations of the reactor radial power distribution. Although thirty plutonium-aluminum alloy elements were used in the initial reactor loading, the number

-180-

of plutonium-aluminum fuel elements required is somewhat arbitrary, since they may be located at the periphery of the reactor core or moved towards the center, thus allowing a variation of about 10 to 12 assemblies added or removed from the loading. This flexibility makes possible large modifications in the power distribution with a minimum of complication and permits latitude in scheduling fuel fabrication.

CALANDRIA

The calandria is a vertical aluminum cylinder with an inside diameter of 84 inches and an inside height of 115 inches. Passing vertically through the calandria are 85 fuel channels, 18 shim control channels and 13 flux monitor channels. Shroud tubes which are welded into calandria tube sheets surround the fuel channels through which the process tubes pass. Eighty-four of the shroud tubes are sized to accommodate standard 3.25 in. O.D. process tubes. The center shroud tube is of a larger size in order to accommodate special experiments. An annular dump chamber encircles the bottom of the calandria and provides initial storage space for a substantial portion of the moderator when it is dumped. An annular gas header encircles the top of the calandria and provides a constant pressure gas reservoir.

MODERATOR AND CONTROL (Figure 4)

The heavy water moderator has a free surface at an annular weir which extends around the base of the calandria inside the dump chamber. The helium atmosphere at this point is maintained at sufficiently high pressure, relative to the pressure at the top of the calandria, to support the moderator at any desired level. Both the helium gas and

the moderator are continually circulated. Gas pressure difference is maintained by a compressor and by-pass valves which constitute the reactor's primary control system. Control of the nuclear reaction, and thus the reactor power level, is achieved by controlling the moderator height (figure 4). Height adjustment is made by adjusting the gas pressure differential between the dump chamber and the top of the moderator. Automatic control, fine and coarse manual control, and normal and emergency shutdown control are provided by specific valves in this system. System stability has proven to be very satisfactory and is adequate to keep fluctuations in moderator level within 1 or 2 mm. Moderator level control has proven practical in PRTR operation since it provides excellent shutdown control and freedom from strong flux perturbations.

Automatic reactor control will be utilized for PRTR startup and normal operation after obtaining additional manual operating experience. Automatic control of the reactor period between 20 and 100 seconds during startup (0-2 MW) and of the operating level from 2-100 MW will be utilized. The automatic controller adjusts a single valve to vary moderator height and thus the nuclear reaction. Fast reactor shutdown is accomplished by equalizing the pressure across the moderator, thus dropping the moderator into the dump chamber and a D_2O storage tank located about 20 feet below the calandria.

SHIM SYSTEM

Eighteen shim assemblies are used to modify radial and axial flux distributions as well as to compensate for xenon reactivity. Each

PRTR CONTROL SYSTEM

FLUX MONITOR

SHIM CONTROL

CONTROL VALVES
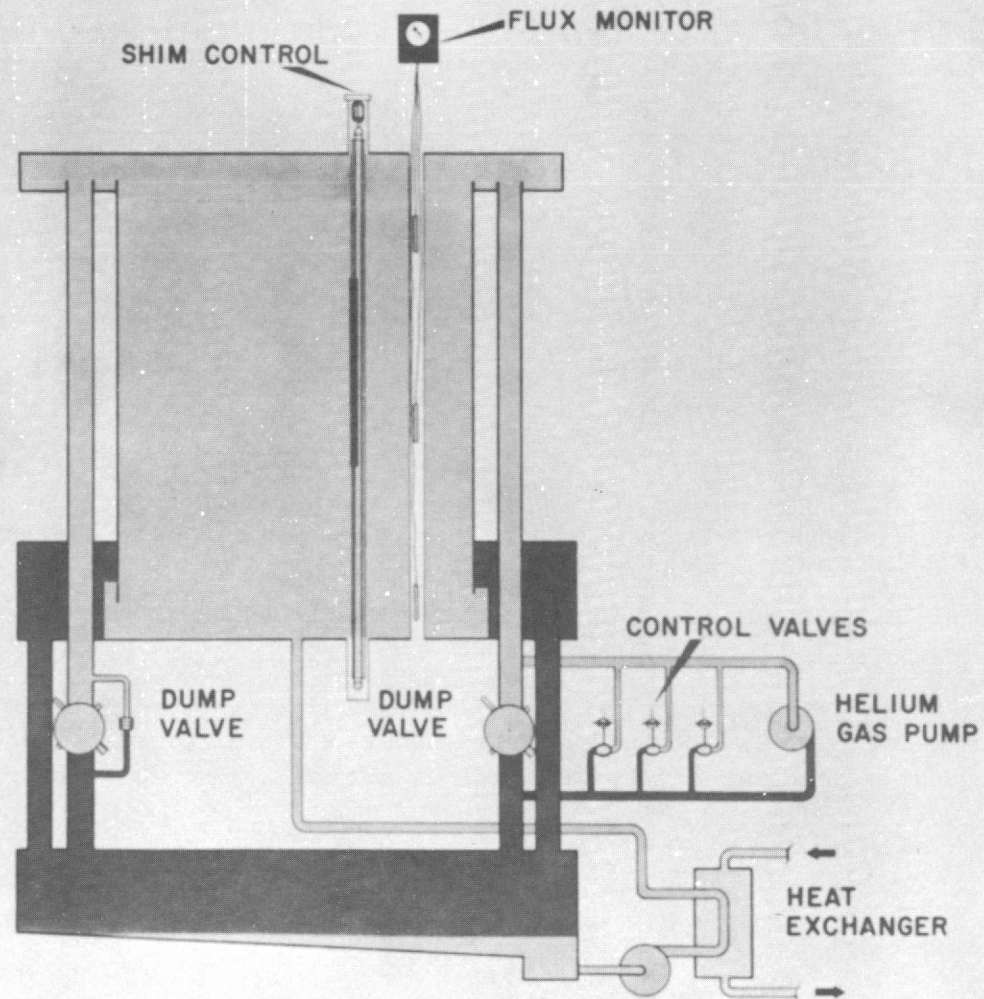
HELIUM GAS PUMP

DUMP VALVE

DUMP VALVE

HEAT EXCHANGER

FIGURE 4

assembly contains three "half-rods", each 35.8 inches in length
and is made of Inconel X. Each rod moves vertically on an aluminum
screw; one rod is driven by an independent motor while a second
motor drives the remaining two rods. Total worth of 54 half-rods is
88 mk, while a half-rod can control 4 mk at most. Rods are re-
cessed into the bottom shield when they are withdrawn from the core
and the motor drives and position transmitters are recessed in the
upper shield. During reactor startup the primary controls are used
only to compensate for temperature coefficients. Xenon compensation
is provided by the shim system.

REFLECTOR

The heavy water reflector is contained in an annular cylinder 11 feet
O.D., 7 feet I.D. and 6 feet 10 inches high. This vessel surrounds
the calandria and is located vertically between the dump chamber and
the gas header. Eight flux chamber holes and two test holes are
spaced at two different levels around the reflector vessel. These
facilities consist of 6-inch O.D. aluminum tubes which pass through
the outer wall of the reflector vessel and terminate at the inner wall.

SHIELDING

A thermal shield, consisting of 12 iron slabs, 6 inches thick,
arranged cylindrically outside of the reflector, is provided to remove
approximately 90 per cent of the radiation energy escaping from the
reflector. Individual thermal blocks are about 3 feet wide and
11 feet high. The thermal shield is cooled by a series of vertical
cooling tubes leaded into grooves on the inside face.

A biological shield surrounds the thermal shield. The biological shield is a cylindrical wall of high density concrete, 71 inches thick and about 21 feet high. The inner face of the biological shield is cooled by circulating water through coils of one-half inch tubing.

The calandria and reactor core components rest on the bottom primary shield. This shield, as well as the top primary shield directly above the calandria, consists of a cylindrical steel tank, 40 inches high, filled with a mixture of iron pellets and water. In addition to attenuating radiation, the water is circulated to remove heat.

The rotating disc shield is located above the top primary shield and consists of two stationary rings which house two rotating discs and an access plug. The central disc is located eccentrically in the outer disc so that the eccentrically located access plug can be positioned over any process tube. The rotating disc shield is constructed of 27 inches of high density concrete. The lower surfaces of the rotating disc shield are cooled by circulating water through tubes embedded in the concrete.

GENERAL (Figures 2 and 5)

The primary coolant inlet ring header, jumpers and associated fittings are located in the lower access room, below the bottom primary shield. Process tube outlet piping is located in the space between the top primary shield and the rotating disc shield. The lower access room opens directly to the process cell. A portable seal, the "thermal
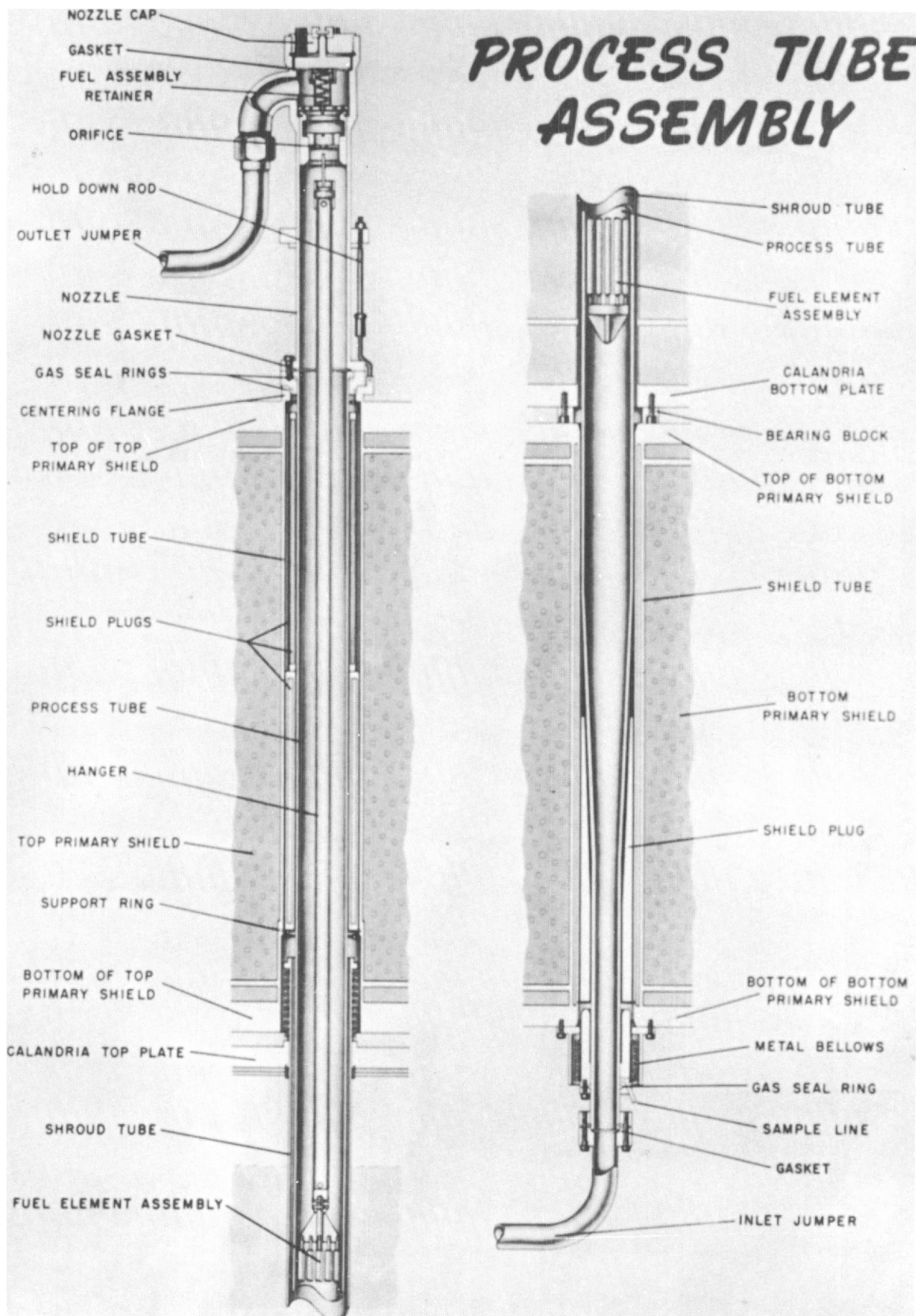
PROCESS TUBE ASSEMBLY

NOZZLE CAP
GASKET
FUEL ASSEMBLY RETAINER
ORIFICE
HOLD DOWN ROD
OUTLET JUMPER
NOZZLE
NOZZLE GASKET
GAS SEAL RINGS
CENTERING FLANGE
TOP OF TOP PRIMARY SHIELD
SHIELD TUBE
SHIELD PLUGS
PROCESS TUBE
HANGER
TOP PRIMARY SHIELD
SUPPORT RING
BOTTOM OF TOP PRIMARY SHIELD
CALANDRIA TOP PLATE
SHROUD TUBE
FUEL ELEMENT ASSEMBLY

SHROUD TUBE
PROCESS TUBE
FUEL ELEMENT ASSEMBLY
CALANDRIA BOTTOM PLATE
BEARING BLOCK
TOP OF BOTTOM PRIMARY SHIELD
SHIELD TUBE
BOTTOM PRIMARY SHIELD
SHIELD PLUG
BOTTOM OF BOTTOM PRIMARY SHIELD
METAL BELLOWS
GAS SEAL RING
SAMPLE LINE
GASKET
INLET JUMPER

FIGURE 5

barrier", separates air flow in the upper region of the lower access room from the lower region of the lower access room and the process cell. Separation of air flow is desirable since air flowing around the process tube fittings may carry D_2O vapor from leakage. Equipment is provided to recover D_2O.

Charge-discharge operations require the use of a self-propelled 50-ton fueling vehicle. This machine is supported by a carriage on a crane-type bridge which travels on floor mounted rails, has a charge tube for loading new fuel elements and an air-cooled discharge cask for discharging irradiated fuel elements, process tubes and shim control rods.

Four top gas lines, four bottom gas lines and a bottom drain line connect the reactor to the moderator storage tank in the bottom of of the lower access room. The top gas lines are provided with quick opening dump valves.

OPERATION

Experience to date has shown that losses of helium and heavy water are only indirectly related to those that one might expect in a power reactor of the same design. Our heavy water loss is thought to be due principally to the large number of mechanical fittings and helium leakage from the primary system, and runs about 1 3/4 per cent per month of the total inventory (or about 1500 pounds D_2O per month). Improved preventive maintenance and methods of leak detection will reduce these losses. We believe the level can be

reduced to about 30 pounds per day. Helium losses have averaged about 150,000 scf (standard cubic feet) per month.

As I have stated earlier, the PRTR went critical on November 21, 1960. The accumulated exposure of the PRTR to date is approximately 8000 MWD's. The time operating efficiency during the past two or three months has been about 50 per cent. The present fuel loading of the reactor consists of about fifty per cent UO_2 and fifty per cent spike elements of Pu-Al. We are in the process of changing the core to a two-zone loading consisting of approximately 13 UO_2, 65 PuO_2-UO_2 and approximately 8 Pu-Al spike elements. This transition is scheduled to be completed by September of this year. The goal exposure of the UO_2 elements now in the reactor is 5000 MWD/TU and the maximum exposure to date is about 2400 MWD/TU. The Pu-Al spike elements have attained exposures of 75 MWD or about 80 full power operating days.

PARTIAL SUMMARY OF TECHNICAL SPECIFICATIONS

I. GENERAL

  A. Reactor

    1. Power Level - 70 MW

      a. Primary Loop 66.25 MW

      b. Remainder of Systems 3.75 MW

    2. Primary Coolant

      a. Flow   8400 gpm at 478 F

      b. Temperature - Inlet   478 F

                    - Outlet (bulk)   530 F

      c. Pressure - Maximum   1125 psig

                    - Minimum   1025 psig

    3. Number of Process Tubes   85

    4. Lattice Spacing   8 in equilateral triangle

    5. Tube Flow - Maximum   123 gpm

    6. Tube Power - Maximum   1200 KW

    7. Number of Shim Rods   54 in 18 channels

    8. Fuel Loading

      a. Minimum Critical Mass - Pu-239 6.64 kg

                       - UO_2 3,323 kg

      b. Excess K, Fresh Loading   13.7 ΔK/K

    9. Flux

      a. Average Thermal Flux 8×10^{13} n/cm^2 sec.

      b. Average Fast Flux 2.6×10^{13} n/cm^2 sec.

   10. Reactivity Coefficients

      a. Moderator Temperature $- 1.06 \times 10^{-4} \ \frac{\Delta K}{K} /C$

      b. Fuel Temperature $- 1.5 \times 10^{-4} \ \frac{\Delta K}{K} /C$

B. Process Area

    1. Containment Vessel

        a. Overall Height   121 ft. 6 in.

        b. Diameter   80 ft.

        c. Wall Thickness

           (1) Dome   9/32 in.

           (2) Shell   9/16 in.

           (3) Bottom   5/8 in.

       d. Material   Carbon Steel

        e. Pressure

           (1) Design Internal   15 psig

           (2) Test Internal   18.75 psig

II. REACTOR COMPONENTS

  A. Fuel Elements

    1. General Design Criteria

        a. Maximum Heat Generation   1200 KW

        b. Maximum Heat Transfer Rate

           at Jacket Surface   400,000 $BTU/hr.ft^2$

    2. 19-Rod Cluster

		Pu-Al	UO_2
a.	Length, Overall	97 5/8 in.	100 5/16 in.
b.	Length, Core Section	87 1/2 in.	88 in.
c.	Diameter, w/wire wrap	3.196 in.	3.196 in.
c.	Weight of Core	33 lbs.	112 lbs.
e.	Weight of Assembly	70 lbs.	138 lbs.
f.	Core Composition (%)	1.8 Pu,95.4 Al	(85%
		1.35 Ni,1.1 Si	Theoretical Density)
g.	Core Diameter	.490 in.	0.502 in.
h.	Cladding	Zircaloy-4	Zircaloy-2

B. **Process Tubes**

    1. Total   85

    2. Material   Zircaloy-2

    3. I.D.   3.250 in.

    4. Wall Thickness   0.159 in.

    5. Length   17 ft. 5 in.

C. **Shroud Tubes**

    1. Total Number   85

    2. Overall Length   10 ft. 5 in.

    3. O.D.

        a. Normal   4.25 in.

        b. Center Shroud Tube   6 in.

        c. Material   Aluminum

D. **Calandria**

    1. Moderator Vessel

        a. Material   Aluminum

        b. Inside Height   114 5/6 in.

        c. I.D.   84 in.

    2. Reflector Vessel

        a. Material   Aluminum

        b. O.D.   11 ft.

        c. I.D.   7 ft.

        d. Height   6 ft. 6 in.

    3. Dump Chamber

        a. Volume   620 gals.

        b. Weir Height   14 3/16 in.

        c. Bottom Gas Lines

            (1) Number   4

            (2) O.D.   10 in.

E. Shielding

    1. Thermal Shield

        a. Material   Iron

        b. Shield Blocks

            (1) Number   12

            (2) Total Weight   10,000 lbs.

        c. Neutron Attenuation 10^{-6}

        d. Shield Temperature   150 F

    2. Biological Shield

        a. Material   Concrete

        b. Density

            (1) Process Cell Side   210 #/ft^3

            (2) Remainder   270 #/ft^3

        c. Neutron Attenuation 10^{-9} to 10^{-10}

        d. Shield Temperature (Inside Face)   120 F

    3. Top and Bottom Shields

        a. Material

            (1) Shield   Carbon Steel

            (2) Shot   Cast Iron

        b. Density   100 lb/ft^3

        c. Maximum Outside Diameter

            (1) Top Shield   14 ft. 10 in.

            (2) Bottom Shield   11 ft. 4 in.

        d. Neutron Attenuation 10^{-6}

        e. Maximum Temperature   150 F

    4. Rotating Disc Shield

        a. Material   Steel Aggregate plus Limonite Grout

        b. Density   320 lb/ft^3

        c. Maximum Temperature   120 F

III. **MAJOR REACTOR SYSTEMS**

    A. <u>Primary Coolant System</u>

        1. General

            a. Heat Removed   66.11 MW

            b. System pH   9.0 - 11.0

            c. Maximum Design Temperature Change 800 F/hr.

        2. Primary Pumps

            a. Number   3

            b. Rating   4200 gpm at 290 ft.

            c. Motor Size   350 HP

        3. Pressurizer

            a. Material   Carbon Steel with 304 SS Cladding

            b. Total Capacity   850 gallons

            c. Pressure

                (1) Operating (Min)   1025 psig

                (2) Design   1250 psig

                (3) Test   1900 psig

            d. Temperature Design   600 F

    B. Secondary Coolant System

        1. Steam Generator

            a. Heat Removed Rate - Normal   65.02 MW

            b. Number of Tubes   1858

            c. Heat Transfer Surface Area

               of Tubes   6850 ft.^{2}

        2. Steam Condenser

            a. Steam Feed   200,000 lbs/hr, 335 F, 5 psig

            b. Effluent   8400 gpm at 125 F

C. <u>Helium System</u>

   1. General

      a. Normal Helium Inventory

         (1) High Pressure Portion   6000 scf

         (2) Low Pressure Portion   2000 scf

      b. High Pressure Storage

         (1) Operating Pressure   2200 psig

      c. Intermediate Pressure Storage

         (1) Operating Pressure   100 - 1000 psig

      d. High Pressure Compressors

         (1) Number   2

         (2) Capacity   55 scfm

         (3) Pressure, Discharge   2200 psig

      e. Low Pressure Compressors

         (1) Number   2

         (2) Capacity   26 scfm

         (3) Pressure, Discharge   170 psig

      f. Gas Balance Compressor

         (1) Capacity   80 scfm

         (2) Pressure, Discharge   3.9 psig

PIQUA NUCLEAR POWER FACILITY AND POPR

by

R. K. Winkleblack

Atomics International

PIQUA

The construction of the Piqua reactor plant has recently been completed. The public hearing was held last week on June 7. The preoperational checking and testing is underway. Sufficient fuel elements have been delivered to the site for cold, clean criticality. It is anticipated that a favorable ruling will be received from the public hearing examiner in four to six weeks and that this will coincide with the completion of the preoperational testing and that fuel loading can begin in the last half of July. A brief description will be given and a number of pictures taken during the construction will be shown. A few of the practical problems encountered will be mentioned. A list of pertinent data is given in the table.

PROTOTYPE ORGANIC POWER REACTOR

The objective of this reactor is to serve as a prototype for reactors in the size range of around 300 Mwe. The Piqua operating conditions were set a few years back and do not utilize the maximum steam cycle capability of the organic coolant. The POPR design is based on uranium oxide fuel contained in finned APM tubes. The core has a hot spot design temperature of 850°F (455°C). This will

enable the plant to operate with a steam pressure of 1000 psi (71 kg/cm^2) and 675°F (357°C). The plant would use a single main coolant loop to obtain experience with as large single components as possible. This would minimize the extrapolation required to design components for plants five to eight times as large as the POPR. The same scheme is proposed for the core. The height of the core would be about 1.7 times greater than the effective diameter; thus, a fuel element would be used which would have the same cross-sectional dimensions as one needed for a larger size plant and only a little shorter.

The following table lists descriptive data for the prototype plant design as well as for the Piqua plant.

REACTOR PLANTS DATA TABLE

	PNPF	POPR
FUEL AND CORE		
Material	U-Mo	UO_2
Cladding	Al	APM (SAP)
Geometry	Concentric Cylinders	Finned Pins
Box Material	S. S.	APM (SAP)
Box Shape	Round	Hexagon
Lattice Arrangement	Spaced	Close Packed
Core Height	4.5 ft	8 ft
Effective Core Diameter	5 ft	4.6 ft
Nominal Hot Spot Temp.	750°F (400°C)	850°F (455°C)
Average Heat Flux, Btu/ft^{2}hr	35,000	48,000
Maximum Heat Flux, Btu/ft^{2}hr	90,000	131,000
Coolant Velocity Max., ft/sec	12.7	20
Core Power Density, kw/liter	18.2	43
Specific Power, kw/kg U	6.6	24
Design Avg. Burnup, MWD/T	3,000	17,500
U-235 Enrichment	1.94%	2.5-3%
COOLANT AND STEAM SYSTEM		
Thermal Power	45.5 Mw	160 Mw
Nominal Operating Organic Press.	120 psig	150 psig
Coolant Design Press.	300 psig	250 psig
Number of Main Coolant Pumps	2	2
Material of Pipes and Vessels	Carbon Steel	Carbon Steel
Number of Boilers	1	1
Number of Superheaters	1	1
Terphenyl Temp. to Superheater	575°F (302°C)	700°F (371°C)
Steam Press. from Superheater	435 psig	1030 psig
Steam Temp. from Superheater	550°F (288°C)	675°F (357°C)
Net Electrical Efficiency	25%	32%

THE EXPERIMENTAL ORGANIC COOLED REACTOR

J. W. Dykes, Phillips Petroleum Company

I. BACKGROUND

Following early operation of OMRE establishing the technical feasibility of an organic cooled and moderated reactor, the U. S. Atomic Energy Commission embarked on an expanded program to explore the potential of the organic reactor for attaining electrical power at competitive costs with conventional plants. The Experimental Organic Cooled Reactor was authorized as a part of the research and development effort to support this program.

The EOCR, together with its experimental facilities, will comprise a versatile tool for experimental investigations in many phases of the organic program.

At the request of the Idaho Operations Office, USAEC, the conceptual design for the reactor was prepared by Phillips Petroleum Company during the latter part of 1959. The facility was designed by the Fluor Corporation with assistance on the nuclear design by Atomics International. The construction contractor for the plant is C. F. Braun Company.

II. DESCRIPTION OF REACTOR & PLANT

A. General

The EOCR was designed to operate at a thermal power of 40 megawatts. The circulating coolant will initially be Santowax OMP and will be pumped through the reactor vessel at a rate of 25,000 gpm, inlet pressure of 160 psig, and inlet temperature of 500°F. The anticipated core pressure drop is 40 psi. The core differential temperature will be 23°F at 40 MW. Heat transferred to the coolant from the reactor core is rejected in an air-cooled, fintube heat exchanger.

Supporting the primary coolant loop are (a) The Coolant Degassing System for removal of undesirable low boiling decomposition products, (b) The Coolant Purification System for removal and control of high-boiling compounds, (c) The Particulate Removal System for removal of particulate matter by distillation, and (d) The Decay Heat Removal System for emergency and shutdown removal of core heat when the main circulating pump is not operating.

The primary loop is constructed of 24 inch carbon steel pipe. All welds in the primary loop were 100 per cent radiographed. The primary loop and auxiliary organic systems will be hydrostatically tested at 570 psig.

The reactor incorporates provisions for five large in-core experimental loops, 16 periphery test element positions, and 12 small capsule locations.

Pressure Vessel & Internals

The pressure vessel is 32 feet in overall length, 95 inches I.D. at the upper section, and 65 inches I.D. at the lower section. The vessel is constructed of Type SA-204 carbon steel. Type 304 stainless cladding on the inside wall above the top spider and on the bottom surface of the top head will minimize corrosion to those surfaces periodically exposed to air.

The upper section of the vessel contains two 24 inch nozzles which serve as inlet and outlet main coolant connections to the vessel. In addition, one 12 inch and one 10 inch nozzle are for exit and inlet lines, respectively, for the decay heat loop. Nine 8 inch nozzles are located just below the top flange to carry instrument leads from experiments out of the reactor vessel. Two 10 inch nozzles, set at 30° angles from the vertical, penetrate the vessel wall just below the top flange providing discharge chutes (communicating with the wash cells) for spent fuel elements and control rods.

An inner tank supports the reactor grid plate and forms a barrier separating inlet and outlet cooling flow. Inlet flow is directed downward through the annulus formed by the inner tank and pressure vessel, then upward through the core and into the exit nozzle. Two concentric 3 inch thick carbon steel thermal shields are located outside the inner tank for protection of the pressure vessel walls from gamma heating. The vessel outer wall is heated by electrical resistance heaters and insulated.

C. Reactor Core, Driver Fuel & Control Rod Assemblies

The EOCR core consists of 20 driver fuel assemblies, 12 control rods, and positions for five major experimental loops. The driver fuel assemblies, control rods, and experimental loop positions are spaced on a square center-to-center lattice of 5.75 inches, providing a separation of 0.5 inches between most assemblies. Sixteen positions ("P" positions) around the core periphery can accommodate prototype power reactor test elements, while twelve smaller positions ("E" positions) are available for capsule irradiations.

Outside the core periphery are two fission chamber thimbles, a source tube, and position for a hydraulic rabbit facility. (Core Figure attached)

The driver fuel assemblies have overall fuel section dimensions of 6.5 x 4 inches, consisting of two rectangular sections each 3 x 4 inches and spaced 0.5 inches apart (Figures attached). It consists of three sub-assemblies: The fuel section, center extension, and top extension. Each of the rectangular fuel sections contains 28 active fuel plates. The complete driver assembly contains 2004 g of U-235. Fuel is fully enriched UO_2 - stainless steel cermet, 0.020 inch thick clad on each side with 0.005 inch type 304 stainless steel. Active length is 36 inches.

The control rod consists of a fuel follower sub-assembly, poison section, and related extension pieces (Figure attached). The fuel follower section is 4 inches square and contains 26 active

fuel plates having a total of 1287 g of U-235. The poison section consists of a square stainless steel box which supports four 0.25 inches thick poison bearing plates. The poison material is a one per cent boron - 10 stainless steel cermet, clad (excluding edges) with stainless steel. The control rod assembly moves as a unit through a close fitting guide boot bolted to the reactor grid plate.

The control rod drive consists of a double Acme screw drive mechanism mounted on the top head. A magnet-energized ball-detent latch holds the rod cocked against a 180 pound scram spring, releasing it upon receipt of scram signal.

D. Unloading Wash Cells

Two wash cells communicate through electrically operated wedgeplug valves to the previously mentioned wash cell nozzles. The lower end of each cell consists of a hydraulic cylinder, unloading ramp, and hydraulic piston. The cells are used to solvent wash organic from the fuel and control rod elements before they are transferred to the canal.

E. The Reactor Control System

The reactor control system is designed to prevent operation under unsafe conditions. System interlocks, alarms, and automatic power reductions are programmed into the control circuitry to provide necessary safety functions. Control rod withdrawal, for example, is restricted at all times to the simultaneous withdrawal of only two rods. Two of the four central rods must be withdrawn before peripheral rods can be withdrawn. Any one of four rods can be selected to act as a regulating rod for automatic control of power level. Power reductions are programmed as SETBACK (regulating rods are driven into the core) and SCRAM (all control rod magnet power is de-energized and all rods drop) depending upon the condition requiring the shutdown.

The nuclear instrumentation provides continuous indication of reactor power by monitoring neutron leakage from the core. This instrumentation is divided into three ranges: source, intermediate (Log N), and power level. These channels cover a span of more than 10 decades. Transistorized circuitry is employed throughout.

III. UTILIZATION OF THE EOCR

The EOCR is a unique research facility. The reactor and plant are experiments in themselves, while at the same time irradiation facilities are made available to conduct supporting experiments under other environmental conditions.

Proposed programs for the EOCR include: (1) further development of organic reactor technology and testing and development of process equipment; (2) performance testing of various organics by investigation of coolant decomposition rates and fouling problems at power reactor conditions; and (3) development and proof-testing low cost fuel assemblies for use in organic power reactors.

Paralleling initial construction of the reactor is construction of two major loop facilities, the Fuel Technology Loop (FTL) and the Coolant Technology Loop (CTL). These loops together with the reactor will contribute to the overall organic research program as follows:

A. Reactor Technology

A reactor capable of operating over a range of power levels, and which will accommodate different core configurations, can yield important results vital to predicting the performance of future reactors in the fields of reactor physics, organic technology, and general plant operation.

An important part of the reactor startup program will be the establishment of operating power. This will involve the use of the Fuel Technology Loop during the shakedown period. The fact that the FTL will be installed prior to power operation provides an opportunity to establish a safe operating power without the likelihood of experiencing excessive plate temperatures on any fuel element in the reactor core. A fuel element in FTL will be exposed to conditions more severe than those existing anywhere in the core proper; in particular, the hot spot on the loop element will be hotter than any core element by a determined amount. Thus, any probability of burnout during the initial approach to power is confined to the element in the FTL. The first phase of the experiment will increase EOCR power until the isolated loop element shows either evidence of nuclear boiling, or fails as evidenced by the release of fission products in the loop.

B. Coolant Technology

Both reactor operation and operation of the Coolant Technology Loop will contribute valuable information toward expanding existing coolant technology. Several characteristics of a coolant are susceptible to study in the EOCR and the CTL. These include:

(1) Rate of decomposition under in-pile conditions.
(2) The nature of decomposition products.
(3) The rate of build-up of high boilers.
(4) Composition under varying process conditions as a function of time.
(5) Physical properties of coolant under in-pile conditions.
(6) Reaction of coolant with surfaces; in particular, the fouling of heat-transfer surfaces.
(7) The effects of additives on any of the above items.

The experiments under this portion of the program are designed to determine relative merits of proposed coolants and to establish standard indices (of decomposition rates, fouling rates, heat transfer properties, etc.) from which other organics can be evaluated.

C. Fuel Technology

Testing of prototype power reactor fuel assemblies can be conducted under a variety of conditions through the availability of regular EOCR core facilities and special loop facilities. Early availability of the Fuel Technology Loop will provide a major facility for these investigations.

IV. PRESENT STATUS OF THE EOCR

Construction of the EOCR is now estimated to be 90 per cent complete. Hydrostatic testing of the vessel and primary piping is currently in progress. Installation of reactor internals is scheduled to commence this month (June 1962). It is anticipated that plant testing will continue through the balance of the year with criticality scheduled early in 1963.

EOCR DATA

Reactor Core

Coolant and Moderator	Santowax OMP
Maximum Design Power, Megawatts	40
Expected Initial Operation Power Range, Megawatts	15-25
Number of Positions in Driver Core:	
Driver fuel assemblies	20
Control-rod fuel followers	12
6.5 in. x 6.5 in. in-core loop positions	5
Mass of U-235 in Initial Core (24 fuel assemblies, 13 fuel followers) kilograms	64.8
Fuel Material	UO_2-SS Cermet

Reactor Physics of Initial Core at 500°F

k_{eff} at 500°F, Hot and Clean:

With 12 control-rod fuel followers	1.07
With 12 control-rod poison sections	0.84

Allocation of Excess Reactivity:

Equilibrium xenon and samarium	0.04
Low-cross-section fission products	0.015
Fuel burnup	0.015

Delayed-Neutron Fraction	0.006738
Prompt-Neutron Lifetime, microseconds	15 to 47

Temperature Coefficient, $\Delta k/°F \times 10^5$:

Reactor Isothermal	-10
Coolant-partial	-11

Coolant Void Coefficient, $\Delta k/\%$ void $\times 10^5$	-150

Average Thermal Flux in Driver Fuel/Mw of Power, Wigner-Wilkins @ 500°F:

Beginning	0.10×10^{13}
End	0.08×10^{13}

Power Density in Driver Fuel, watts/cc/Mw of Reactor Power:

Average:	Beginning of core life		2.2
	End of core life		2.6
Maximum			10.0

Vertical Max-Avg Power Ratio   1.4

Reactor Heat Transfer

Heat Transfer Area of 24 Fuel Assemblies and
12 Fuel Followers, sq ft   2146

Velocity of Coolant in Orificed Fuel Assemblies and
Fuel Followers, ft/sec:

Maximum	18.8
Average	12.6
Minimum	7.9

Nominal Reactor Pressure, psig   150

Core Pressure-Drop, psi   40

Inlet Coolant Temperature, $^{\circ}$F   500

Outlet Coolant Temperature at 40 Megawatts Power, $^{\circ}$F   523

Nominal Heat Flux/Mw Reactor Power, Btu ft^{-2} hr^{-1}:

Average:	Beginning of core life	1900
	End of core life	1590
Maximum (in position D-11)		7300

Burnout Heat Flux with 5% HB, Btu ft^{-2} hr^{-1} 1.0×10^{6}

Process Systems

Total Coolant Flow Rate, gpm   25,000

Total System Pressure-Drop, psi   85

Number of Primary Coolant Pumps   1

Heat-Removal Capacity with 500°F Coolant, Megawatts   40

Design Temperature, $^{\circ}$F   850

Design Pressure, psig   335

EOCR   REACTOR
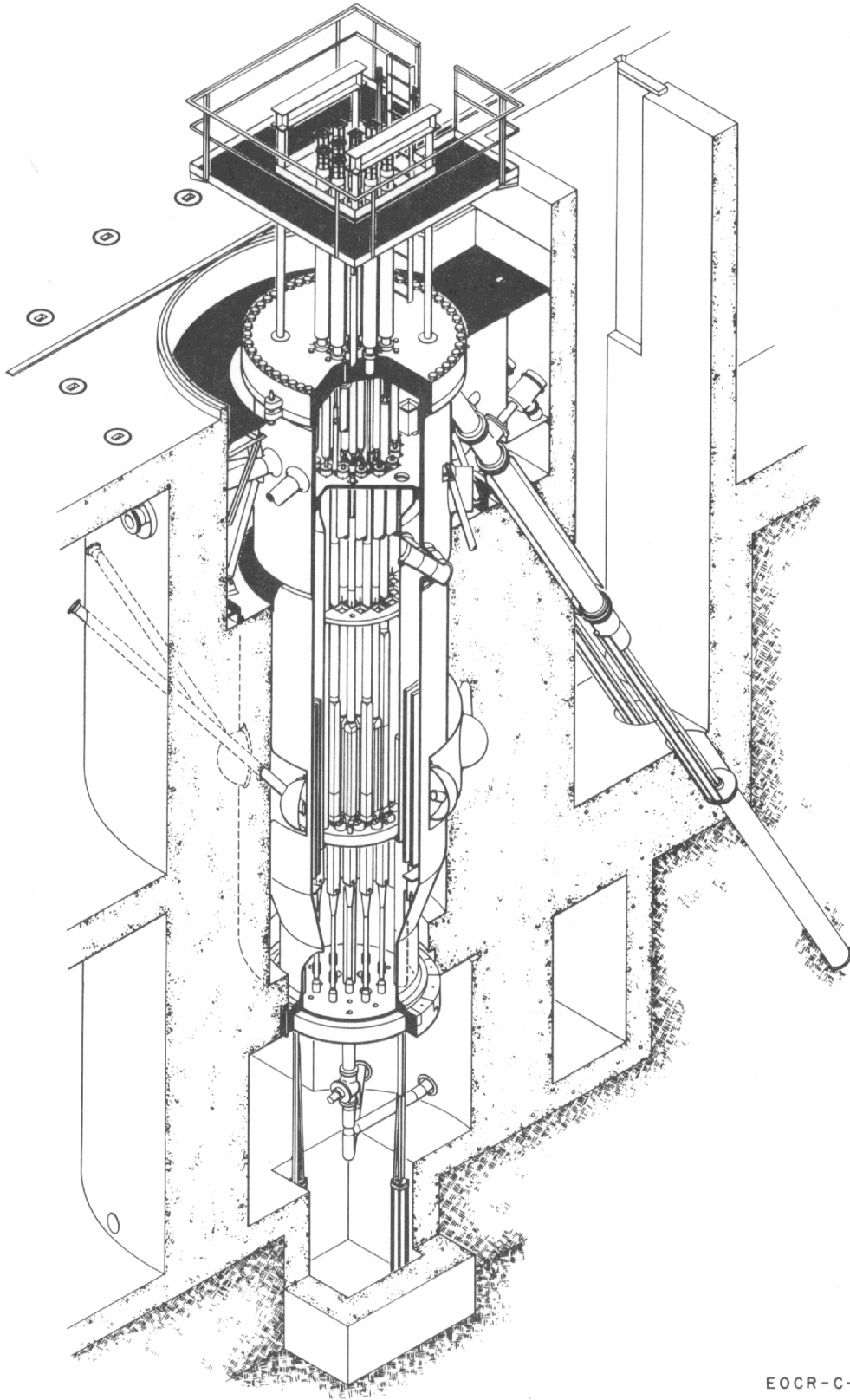
EOCR-C-01

Fig. ii-7 EOCR Isometric View

FUEL HANDLING
BRIDGE

FUEL HANDLING
EXTENSION TANK

LEAD SHIELDING
RING

SHIELDING
BLOCKS

CANAL

SPIDER
No. 1

DISCHARGE
VALVE

SPIDER
No. 2

WASH
CELL

ION-CHAMBER
THIMBLE

CORE

GRID

SUB-PILE
ROOM

RE-ENTRANT
EXPERIMENTAL
LOOP

HYDRAULIC
CYLINDER

EOCR-B-096
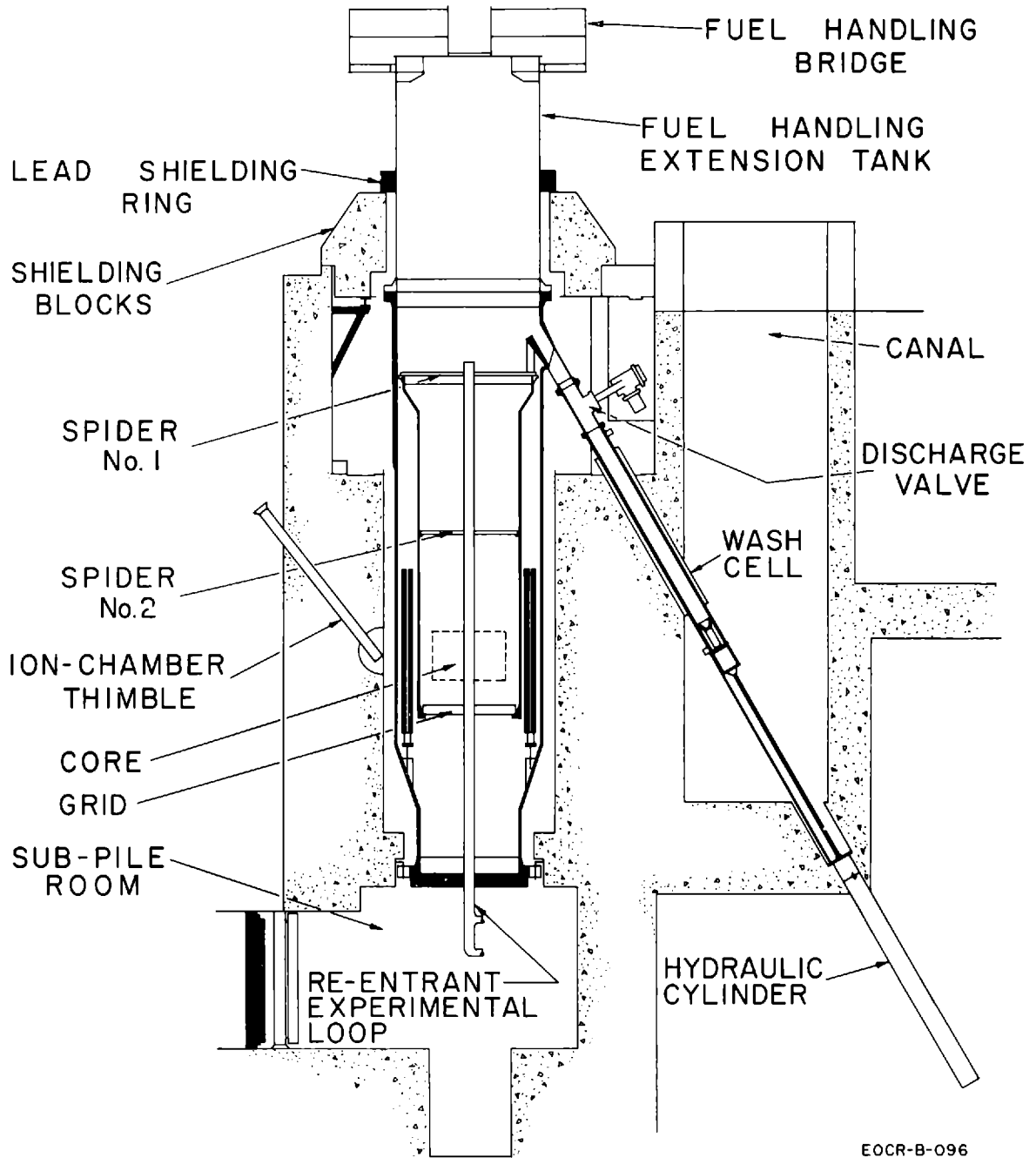
Fig. iii-11   Reactor and Canal Elevation - Functional Layout

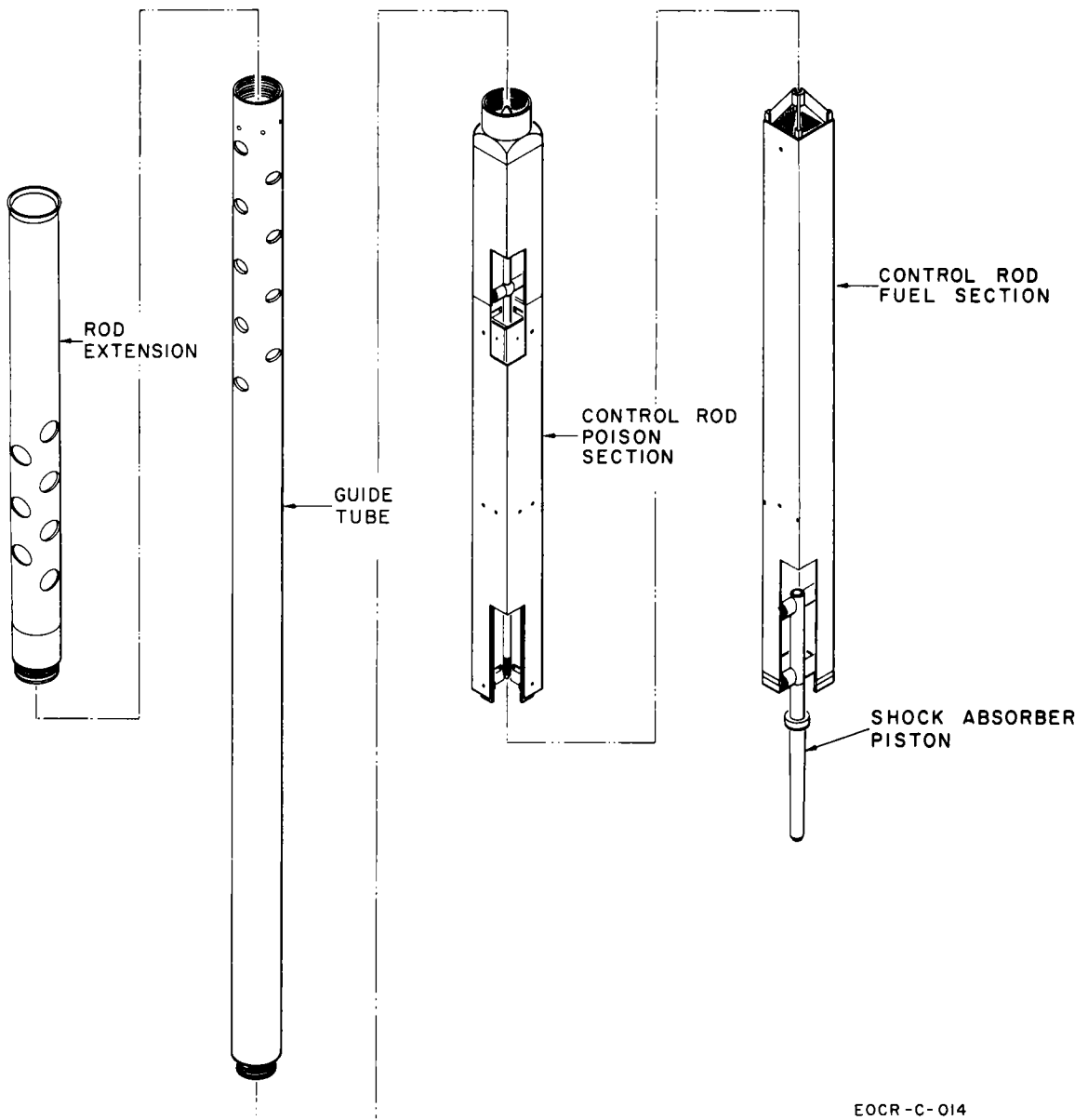Fig. iii-2 Core Layout

EOCR DRIVER
FUEL
ASSEMBLY

TOP
EXTENSION

CENTER
EXTENSION

A
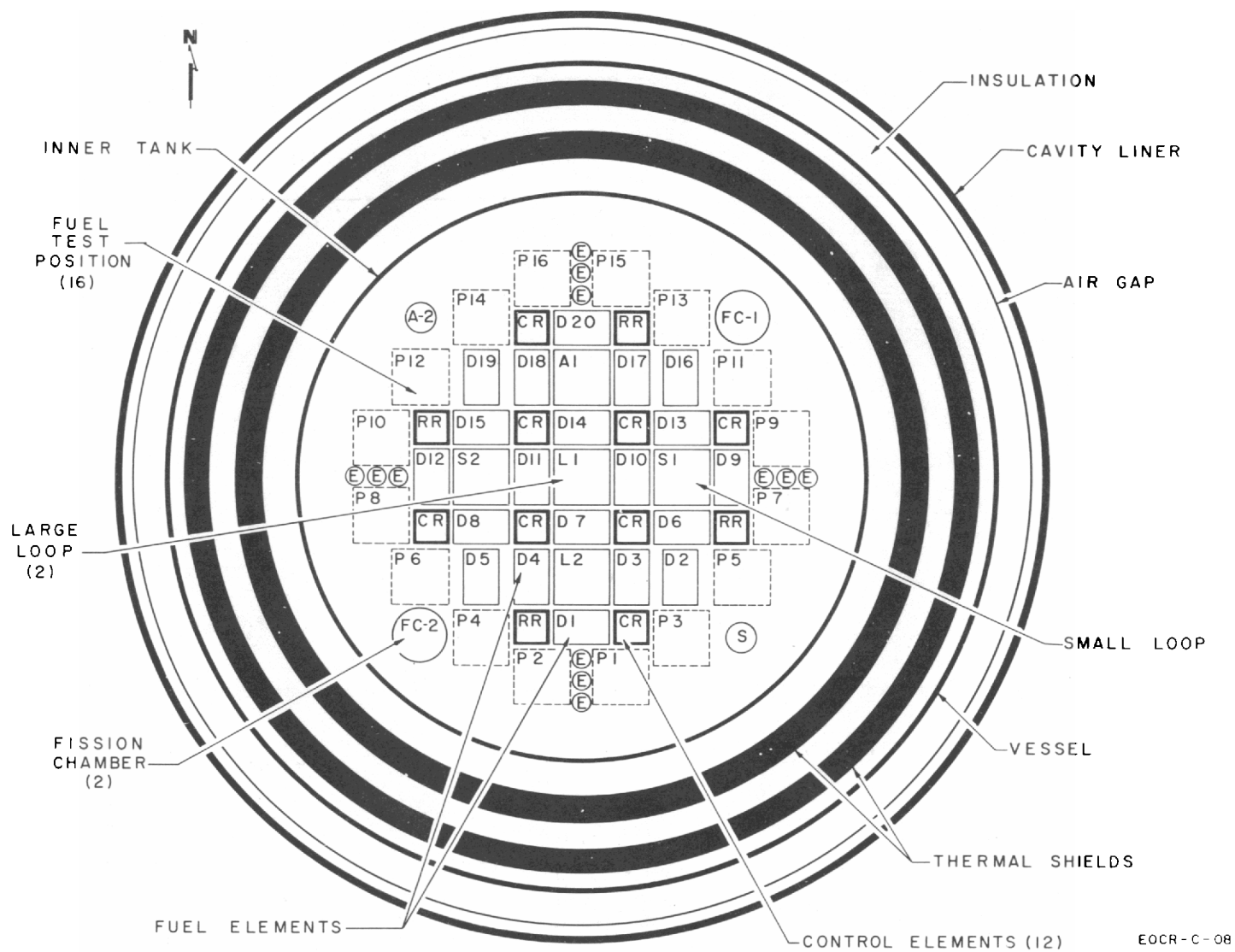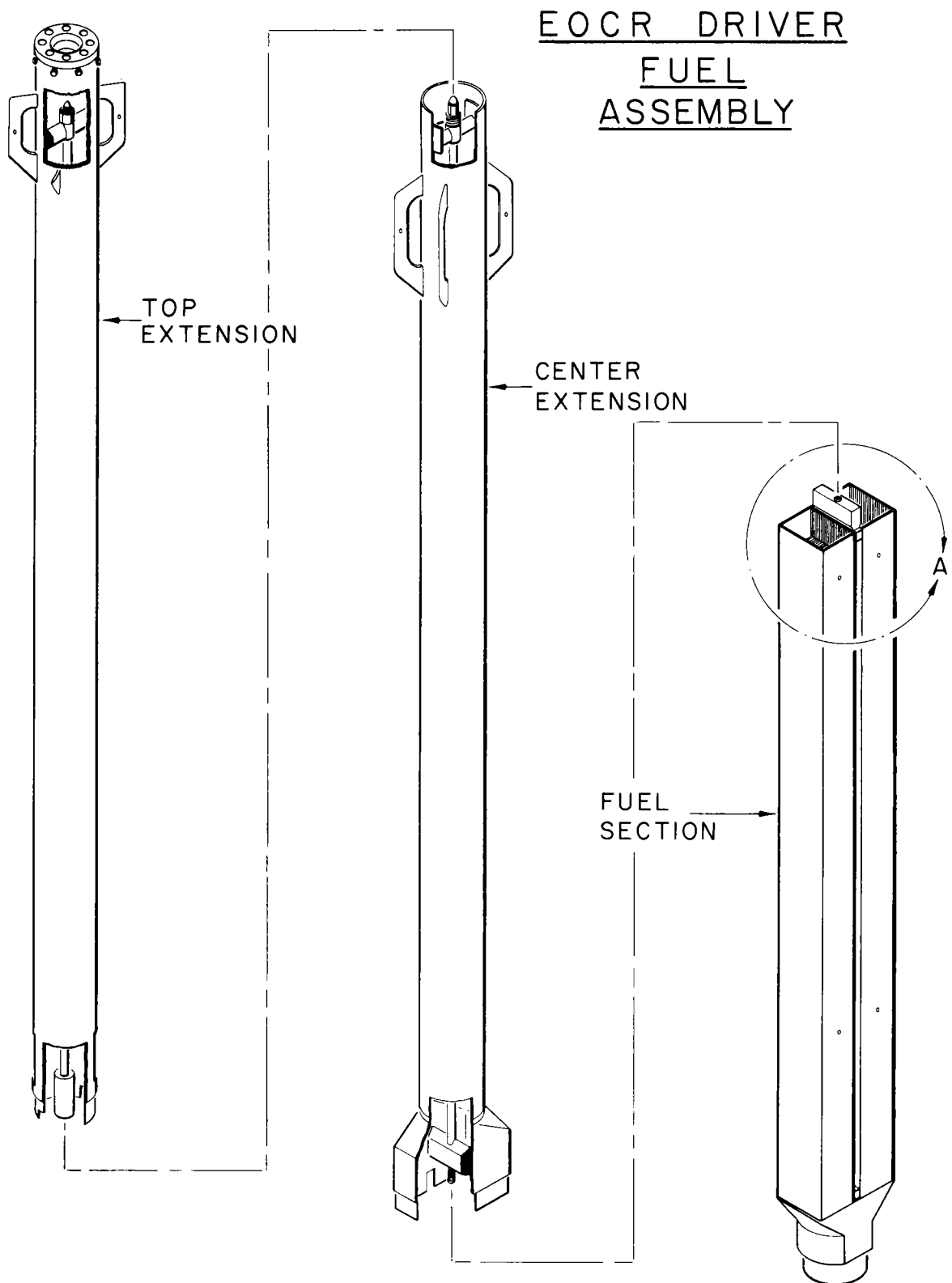
FUEL
SECTION

Fig. iii-5a Driver Fuel Assembly Components

±0.002
OVERALL FUEL PLATE WIDTH 2.844
FUEL PLATE MEAT WIDTH NOMINAL 2.404

29 PLATES (28 ACTIVE PLATES)
30 COOLANT CHANNELS

DUMMY SUPPORT PLATE

38.5
FUEL LENGTH
35.0 MIN.
36.0 NOM.
36.5 MAX.

±0.002
0.070
0.100
0.0977 (TYP)
0.020 FUEL
0.030
0.005 CLAD.
0.140 ±0.040
0.080 ±0.010

VIEW "A"

EOCR-B-026
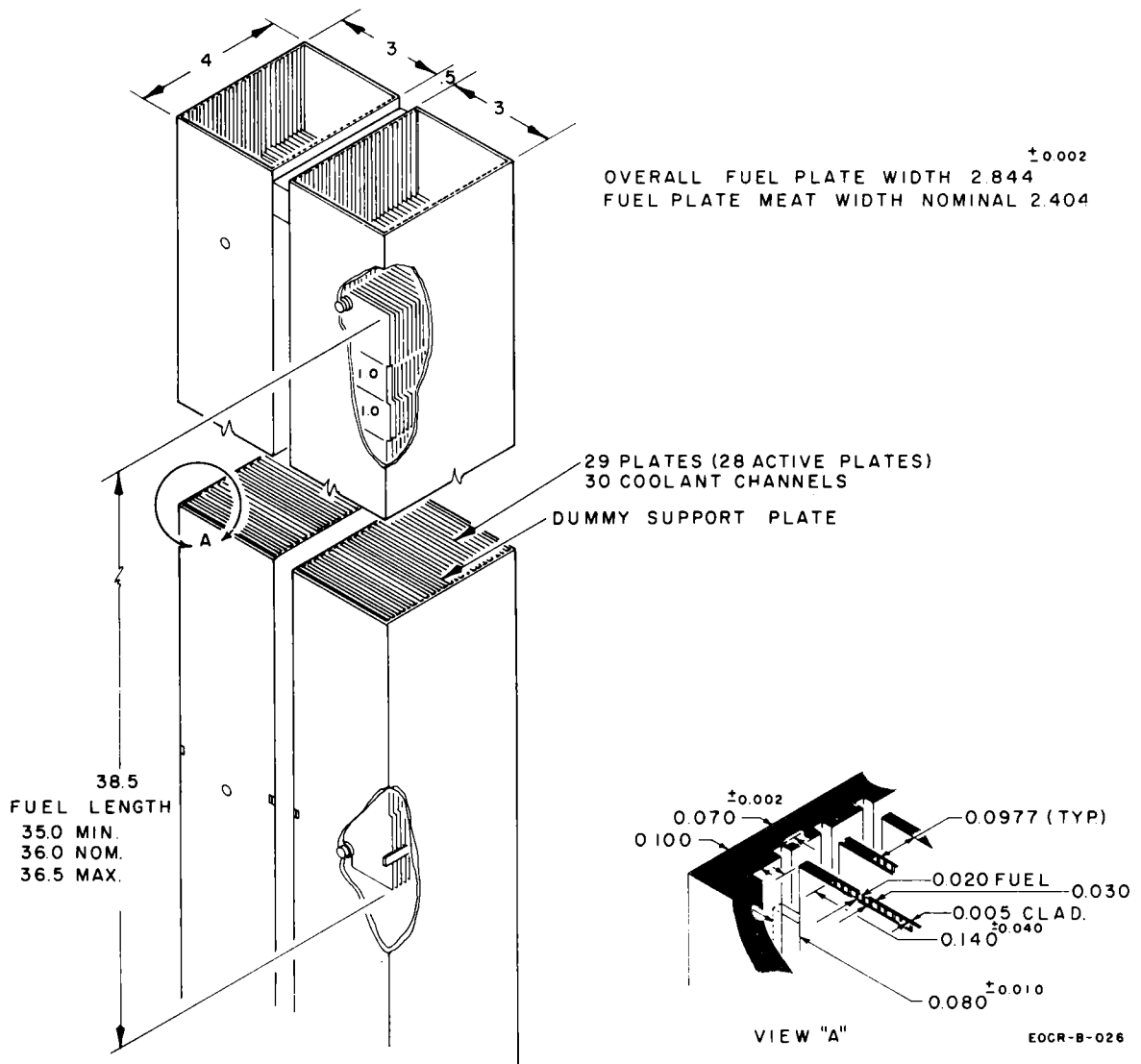
Fig. iii-5b Driver Fuel Assembly Details

EOCR CONTROL ROD SUBASSEMBLIES

ROD
EXTENSION

GUIDE
TUBE

CONTROL ROD
POISON
SECTION

CONTROL ROD
FUEL SECTION

SHOCK ABSORBER
PISTON

EOCR-C-014

Fig. iii-6a Control Rod Sub-Assemblies
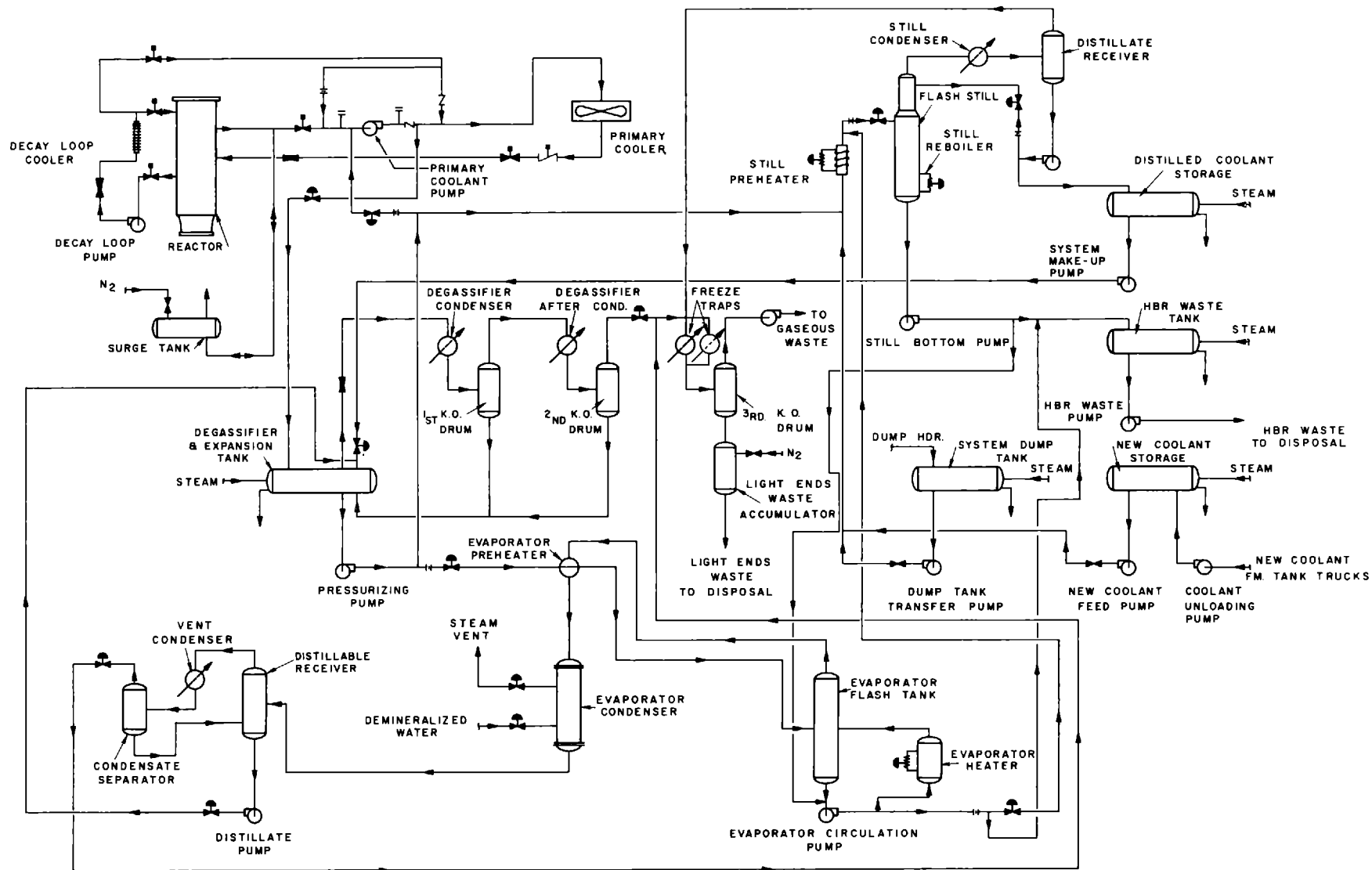
Fig. ii-8 Reactor Coolant System

EOCR-C-0102

A COMPUTER PROGRAM FOR ECONOMIC STUDIES *
OF HEAVY WATER POWER REACTORS

D. S. St. John

The du Pont Company is conducting a study for the United States
Atomic Energy Commission to establish the technology required for
successful design, construction and operation of heavy-water-moderated
natural-uranium-fueled power reactors that would compete with large-
scale fossil fueled power plants. To assist in this study, du Pont
in cooperation with Sargent and Lundy, Engineers, has developed a
computer code that can evaluate rapidly the design and cost of a D_2O
power reactor system as a function of various engineering and
economic criteria.

The code as prepared initially is designed to study 300 MWe reactors
cooled with D_2O. These reactors can be cooled with liquid or boiling
D_2O and fueled with natural or slightly enriched uranium in the form
of metal or UO_2 fuel assemblies consisting of two or three nested
tubes. Modifications are currently underway to adapt the code for
organic coolant and uranium carbide fuel. Sargent and Lundy is
preparing the turbine plant data required to adapt the code for
reactors with 500 and 1000 MWe output.

1. Description of the Code

The computer code starts with a minimum but sufficient set
of parameters describing the fuel assembly, the reactor, and
the required power output. The detailed physics and engineer-
ing calculations required to establish the fuel assembly
dimensions and the design of the reactor system, the steam
turbine plant, and all of the auxiliaries are performed. The
investment required for equipment, materials, buildings and
facilities are computed and all of the cost accounting cal-
culations are carried out that are required to obtain the
total cost of the energy from the designed plant in mills/kwh.

The main features of the code are described in Reference 1 and
the equations and specific costs used in the calculations are
given in Reference 2.

---

2. Use of the Code

   The computer code is an invaluable tool for carrying out the
   large number of calculations required to determine the effect
   of a change in engineering limits or economic groundrules on
   the cost of power obtained from the reactor system designed
   to be optimum under the imposed limitations. The code can be
   used (a) to determine the preferred conceptual design, (b) to
   compare the system economically with fossil-fueled power
   stations or other nuclear plants, (c) to determine the future
   economic potential, (d) to study the effects of varying economic
   conditions, and (e) to determine the economic value of a
   successful research and development program to relax one of the
   engineering limitations.

3. Modifications Required for Organic Coolant

   The code is currently being modified to accommodate organic-
   cooled designs. The modifications include the hydraulic, heat
   transfer, and physics parameters necessary to accommodate organic
   coolant and uranium carbide fuel, an extra iteration step to
   determine the optimum degree of superheat, and facilities for
   maintaining organic quality.

4. Preliminary Results

   Before the decision was made to modify the code to include organic
   coolant, an initial economic comparison was made among organic-
   cooled, gas-cooled. and D_2O-cooled concepts. The results of this
   initial study showed that the organic cooled concept was one of
   the most promising and might have capital costs 10% below D_2O-
   cooled designs. The major uncertainty in this comparison
   was the ability of a uranium carbide and organic fuel assembly
   of natural uranium to provide the reactivity required to give
   economically attractive power distributions and fuel exposures.
   The codes for calculating the reactor physics and the core design
   have now been completed and checks with experimental measurements
   of the reactivity have been made. The results of the preliminary
   survey with these codes is that we believe it possible to have
   economical fuel and reactor designs with natural uranium carbide
   fuel in tubular fuel assemblies that can achieve more than
   6000 MWD/tonne exposure and still have heat transfer and power
   distribution characteristics that will lead to economically
   attractive systems.

5. Problems and Program

   Some of the remaining problems that are being considered at
   the Savannah River Laboratory are the following:

   (a) Fuel Fabrication and Performance

       The production of long tubes of uranium carbide fuel in
       APM cladding to give a satisfactory and economical fuel
       piece has yet to be demonstrated.

   (b) Initial Reactivity

       Measurements have been made at the Savannah River
       Laboratory of the reactivity of nested uranium metal

and uranium oxide fuel pieces with organic material in the coolant channels. Calculations are being made with the THERMOS code to establish the neutron temperature and effective cross sections of the fuel, cladding, and housing materials.

(c) Reactivity Transients

The difference in reactivity transients between organic-cooled and D_2O-cooled reactors is primarily a function of the effective neutron temperature in the fuel. The neutron temperature studies mentioned above are being applied.

(d) Coolant Temperature Coefficient and Stability

The organic-cooled tubular assemblies have a positive coolant temperature coefficient. Estimates are being made of the magnitude of this coefficient from the THERMOS calculations. These will be used together with the negative fuel temperature coefficient in a stability and safety analysis of the reactor system.

6. Conclusion

We expect to be able to use the computer code to establish the economics of organic-cooled reactors on a basis comparable to that of D_2O-cooled reactors by the end of this year. Preliminary surveys indicate that the higher thermal efficiency associated with the organic coolant gives it an economic advantage over the D_2O-cooled reactors. How well this advantage can be maintained in the face of lower reactivity and hence poorer power distribution and shorter fuel exposures will be studied with the computer code.

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FUEL ASSEMBLY DESIGN PROGRAM, FAD*

by

J. C. Jensen

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FAD is a computer program written in FORTRAN for the IBM 704 to design tubular fuel assemblies for heavy-water-moderated reactors. FAD was developed for use with a computer program for the economic optimization of heavy water reactors[1]. This program calculates rapidly a large number of fuel assembly designs, so that the cost of power can be studied as a function of the fuel design parameters.

The basic fuel assembly model in the FAD program consists of nested tubes enclosed by an insulating gap and a pressure tube. The assembly may contain two or three concentric tubes with or without an inner housing tube. Two pressure tube designs may be used as shown in Figure 1. In the "calandria" design shown in Figure 1a the moderator is insulated from the coolant by an annulus of air between the "calandria" and the outside of the pressure tube. In the "non-calandria" design the moderator is insulated from the coolant by an annulus of stagnant D_2O between the pressure tube and an insulating liner. In the latter design the moderator is in contact with the outside of the pressure tube.

The fuel material may be uranium metal or uranium oxide, either enriched or natural. The cladding, pressure tube, insulating liner, calandria, and inner housing tube may be either

aluminum or Zircaloy. The coolant is liquid D_2O. Liquid H_2O or organic coolant may be used by changing the coolant physical property tables.

The assembly is designed to meet three engineering criteria: 1) equal bulk coolant temperatures in all subchannels, 2) equal pressure drop in all subchannels, and 3) equal burnout safety factor in all subchannels. The power output of the assembly is determined from limiting values of the surface temperature, fuel temperature, and burnout safety factor, all of which must be specified for each problem. In addition, the lattice physics parameters are computed for a unit cell.

The computer program is divided into several subroutines as shown by the flow diagram in Figure 2. The main portion of the program is used to handle the input and output data, adjust the power level of the assembly, and control the flow of logic between the various subroutines.

The DESYN subroutine is used to compute the dimensions of the assembly. The dimensions are systematically adjusted by a Newton's method until the engineering criteria are satisfied. The flow and velocity for each coolant channel and the power for each fuel piece are calculated at a given power level for use in the DOTT[2] subroutine.

The P3BSQ[3] subroutine is used to obtain the radial neutron flux distribution within the assembly and the lattice physics constants for a unit cell in an infinite lattice. A single-group P-3 calculation is used to obtain the thermal flux distribution, the thermal utilization, f, and the thermal diffusion area, L^2. The remaining lattice physics parameters are calculated in P3BSQ by methods that have been developed at

the Savannah River Laboratory and have been normalized to experiments conducted in heavy-water-moderated lattices.

The DOTT subroutine is used to calculate the coolant and fuel temperatures within the assembly, the heat split within each fuel piece, and the burnout safety factor. The temperatures and burnout safety factor are calculated at fifty evenly spaced points along the length of the assembly. The maximum surface and fuel temperatures and the minimum burnout safety factor are then obtained from a search of all the previously calculated values. The DOTT program as described in reference 2 has been modified to include the thermal conductivity of the fuel as a function of temperature and a calculation of the burnout safety factor.

Table I shows the input required for operation of the program and includes: weight of fuel, outer coolant channel thickness, type of lattice, lattice pitch, cladding thickness, materials used, reactor inlet and outlet temperatures, and engineering limits. The "Initialize" portion of the program reads and edits the input parameters and sets up all available input parameters for each subroutine. Since the dimensions of the assembly depend on the radial neutron flux distribution and the heat split in each fuel piece, initial guesses for these values are set up as input to the DESYN subroutine.

In the DESYN subroutine, initial dimensions of the assembly are obtained for an assumed power output. Improved values of the radial neutron flux distribution are obtained in the P3BSQ subroutine and new dimensions are calculated.

The assembly temperatures, burnout safety factor, and heat split in each fuel piece are calculated in the DOTT subroutine. The power output of the assembly is then adjusted toward the maximum value as determined from the specified

engineering limits. New assembly dimensions are obtained in the DESYN subroutine with the new power level and the improved values of the heat split in each fuel piece.

When the maximum power of the assembly is obtained, the radial neutron flux distribution within the assembly is again calculated. The design process then repeats until the radial neutron flux has converged.

Some typical results obtained from the FAD program are shown in Figure 3. In these calculations the mass of UO_2 fuel was varied from 17.5 to 30 lb/ft for a fuel length of 15 feet and a lattice pitch of 11 inches. The power output of the assembly was limited in each case by a constant maximum fuel temperature. The specific power of the lattice decreases as the fuel mass, i.e., fuel thickness, is increased, since the power of the assembly is limited by the maximum fuel temperature. The lattice buckling passes through a maximum value as the initially over-moderated lattice becomes under-moderated when the fuel mass is increased from 17.5 to 30 lb/ft.

These results have been used to study the variation in energy costs for a large power reactor as a function of the fuel mass. The minimum in the energy cost results primarily from the competing effects of the increased buckling and decreased specific power which are related to the fuel exposure and the size and costs of the reactor.
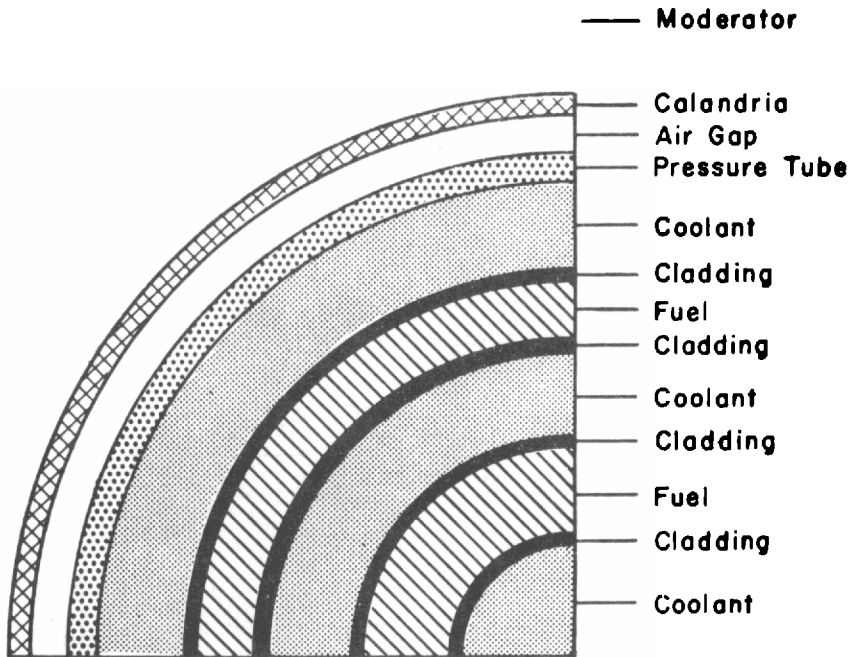
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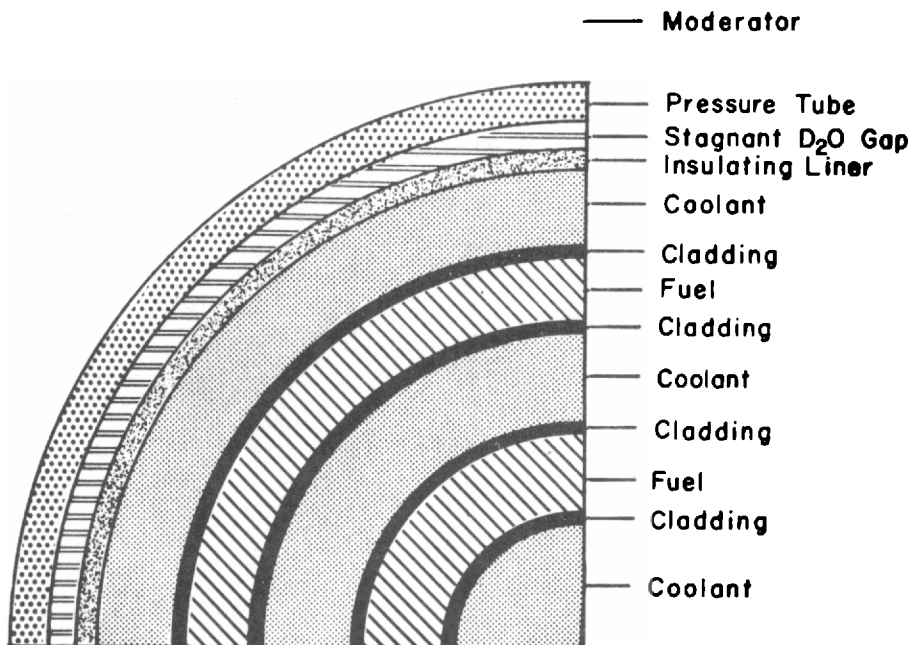
TABLE I

INPUT PARAMETERS FOR FAD

1. Lattice Pitch, inches

2. Lattice Pattern, Hexagonal or Square

3. Coolant Inlet Temperature, °C

4. Coolant Outlet Temperature, °C

5. Moderator Temperature, °C

6. H_2O in Moderator, atom %

7. Fuel Material, UO_2 or U Metal

8. Fuel Length, inches

9. Mass of Fuel per Unit Length, lb/ft

10. Number of Concentric Tubes, 2 or 3

11. Cladding Material, Zr or Al

12. Fuel Density (for UO_2 only), % of theoretical

13. U^{235} Content in Fuel, weight %

14. Thickness of Outer Coolant Channel, mils

15. Insulating Liner or Calandria Tube Material, Zr or Al

16. Thickness of Insulating Annulus, mils

17. Thickness of Inner Housing Tube, mils

18. Cladding Thickness, mils

19. Pressure Tube Material, Al or Zr

20. Maximum Allowable Surface Temperature of Cladding, °C

21. Maximum Allowable Fuel Temperature, °C

22. Minimum Allowable Burnout Safety Factor

FIGURE I
TYPICAL FUEL ASSEMBLIES

— Moderator

Calandria
Air Gap
Pressure Tube

Coolant

Cladding

Fuel

Cladding

Coolant

Cladding

Fuel

Cladding

Coolant

(a) Calandria Type Fuel Assembly

— Moderator

Pressure Tube
Stagnant D_2O Gap
Insulating Liner

Coolant

Cladding
Fuel
Cladding

Coolant

Cladding

Fuel
Cladding

Coolant

(b) Non-Calandria Type Fuel Assembly
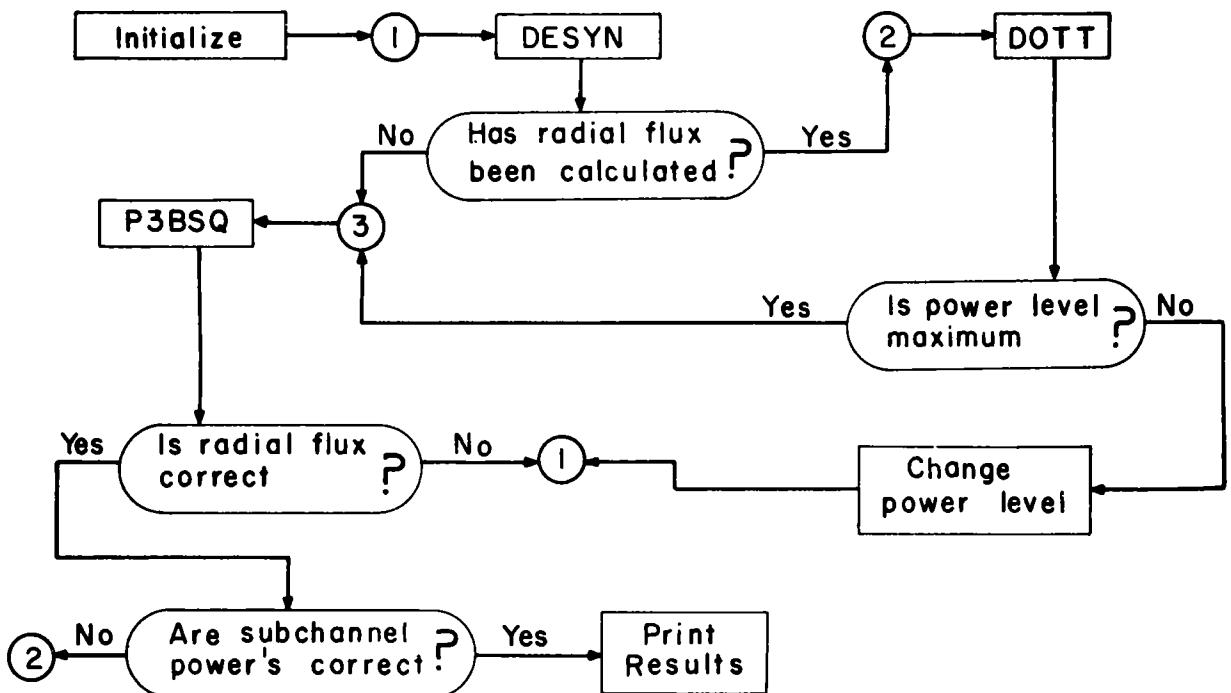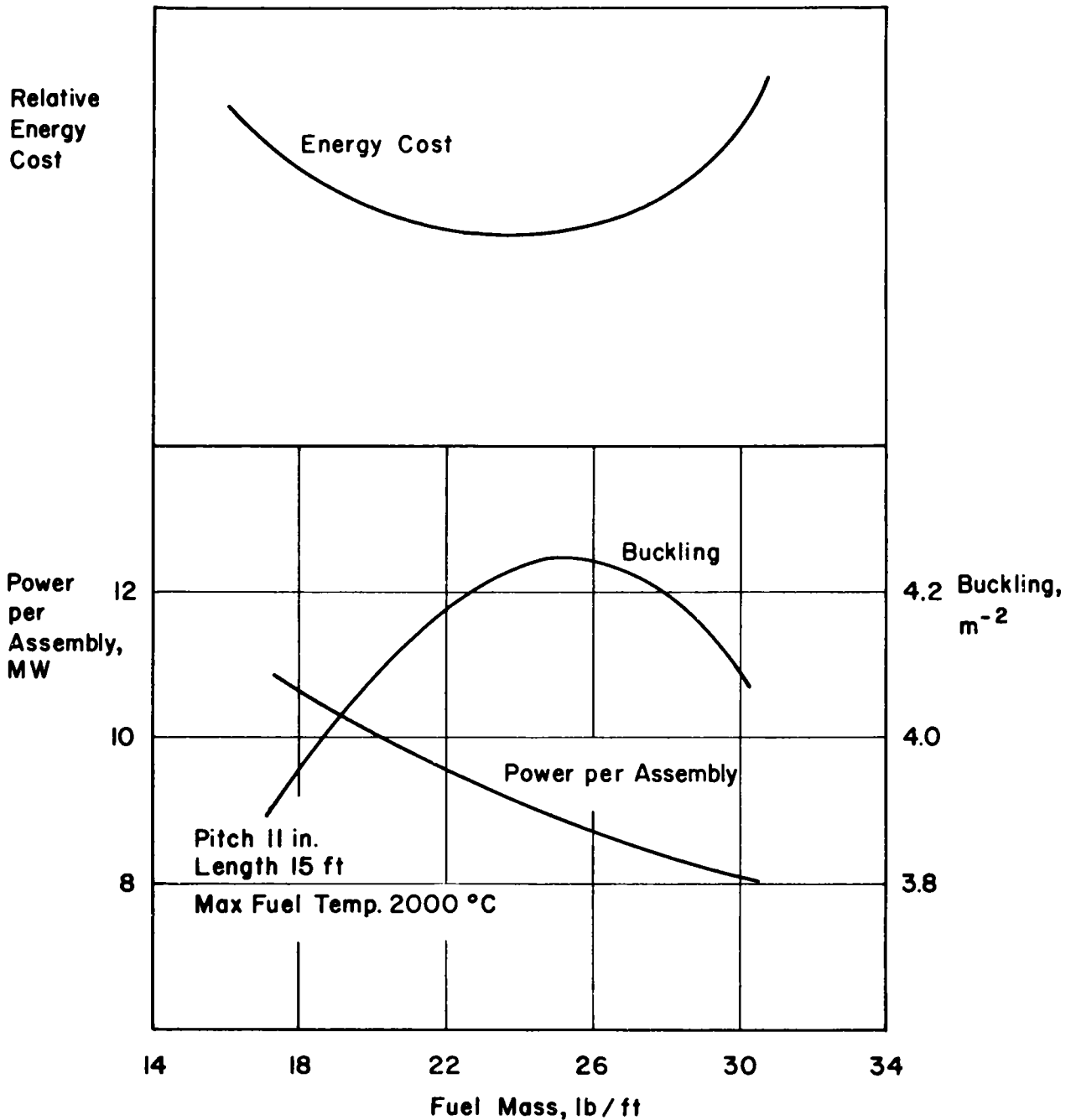
FIGURE 2

FLOWSHEET FOR "FAD" CODE

FIGURE 3

EFFECT OF FUEL MASS ON FUEL
ASSEMBLY POWER AND REACTIVITY

OPTIMIZATION OF HEAVY WATER POWER REACTORS[*]

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Lawrence Isakoff[†], James W. Wade[†]

The United States Atomic Energy Commission, in its program
to promote the use of nuclear energy for peacetime purposes, assigned
Du Pont the task of establishing the technology required for success-
ful design, construction and operation of heavy-water-moderated
natural-uranium-fueled power reactors that would compete with
large-scale fossil-fueled power plants. Among the important problems
facing the developers of these power reactors are the following:

(1) What is the preferred conceptual design for such
a system?

(2) Can this reactor plant compete economically with
fossil-fueled power stations or other types of
nuclear-fueled plants?

(3) What is the future economic potential of this
reactor concept?

(4) What is the economic climate that favors this
type of power plant over others?

(5) What research and development effort is required
to establish the feasibility of D_2O power reactor
designs that are more economical than those that
could be built today?

To assist in answering these questions, Du Pont in cooperation
with Sargent & Lundy, Engineers, developed a computer code that
would rapidly evaluate the design and cost of a D_2O power reactor
system as a function of various engineering and economic criteria.
The purpose of this paper is to illustrate the value of this code

in its application to the above problems. A brief description of the code, its versatility, and the magnitude of the job it performs will be followed by several illustrative examples.

Description of Code

The "Cost Optimization Code" starts with a minimum, but sufficient, set of parameters describing the fuel assembly, the reactor, and the required power output. The detailed physics and engineering calculations required to establish the design of the reactor system, the steam turbine plant, and all the auxiliaries are then performed. Computation of the investment required for the equipment, materials, buildings and facilities follows. Finally, the computer proceeds through all the cost accounting calculations necessary to obtain the total cost of energy from the designed plant in mills per kilowatt-hour.

At present, three types of heavy-water-moderated power plants can be handled by the code. These three concepts, which were chosen from earlier evaluations as suitable for early exploitation,[1,2,3] are as follows:

      (1)   Uranium-metal fueled and liquid-D_2O cooled

      (2)   Uranium-oxide fueled and liquid-D_2O cooled

      (3)   Uranium-oxide fueled and boiling-D_2O cooled

Each of these D_2O-moderated reactors is designed to produce 300 electrical megawatts. Each is a pressure tube reactor; that is, the fuel and the high-temperature D_2O coolant are contained in pressure tubes that pass through the unpressurized liquid D_2O moderator, thereby maintaining the coolant and moderator systems separate. An indirect cycle is employed with the liquid-D_2O-cooled reactors; the fission heat is transferred from the liquid D_2O coolant in a steam generator to produce saturated H_2O steam, which is then delivered to the turbine. A direct cycle is employed with the boiling-D_2O-cooled reactors; the saturated D_2O steam that is generated in the reactor is delivered directly to the turbine plant.

Figure 1 presents a simplified flow diagram of the computer code along with the more important input and output information for each of the major sections, or "blocks". Approximately eight major parameters and about fifty minor ones are required input for each problem.

In the first block of the code, "Fuel Assembly Design," the dimensions of a fuel assembly of multiple coaxial tubes are designed automatically in accordance with several criteria. These criteria involve equalization of each of the following variables in each coolant subchannel of the fuel assembly: (1) pressure drop, (2) coolant temperature, (3) maximum surface temperature, and (4) minimum heat transfer burnout safety factor. The maximum power output of an assembly meeting these criteria is obtained when a limiting temperature, fixed by materials considerations, or a minimum burnout safety factor, fixed by operating considerations, is reached. As an alternative to automatic design of the fuel assembly, any fixed design can be entered into the computations at the end of this code block. Thus, it has been possible to perform complete cost analyses on clustered rod assemblies and other fuel assemblies designed externally to satisfy other engineering or metallurgical criteria. The physics parameters associated with an infinite lattice of fuel assemblies is also computed in this first block of the code.

In the next major code block, "Core Design," the reactor dimensions, the fuel exposure, and the annual fuel throughput are calculated from the total desired thermal output of the reactor, the previously determined heat transfer characteristics of the fuel assembly, and the previously determined nuclear parameters of the infinite lattice. This block of the code considers different fuel charging and power flattening schemes. In these computations, flattening of the power distribution is obtained by charging the fuel in the center of the reactor at a rate such that the mean radial buckling is zero and by adjusting the charging rate in the outer region so that criticality is maintained. In this

block, full-length fuel assemblies are assumed to be charged continuously but at different rates in the central and outer regions.

In the "Fuel Fabrication" block of the code, the costs of manufacturing the required reactor assemblies are computed from the dimensions of the fuel assembly and the annual fuel throughput. The equations relating these parameters to the fuel fabrication cost were derived on the basis of the construction, amortization, and operation of a fabrication plant designed to furnish fuel assemblies for this one reactor at the required rate.[3]

The "Reactor and Auxiliaries" block then calculates the dimensions and costs of the various pieces of reactor equipment, shielding, and auxiliary services and equipment from information supplied by the two preceding blocks. Within the "Reactor and Auxiliaries" block, a minimum-cost containment vessel is automatically designed. This vessel is designed according to ASME code specifications to withstand the maximum pressure buildup that might result from the release of all the coolant in the primary system. The equations relating the size, pressure, and temperature of the various components of the system to their costs were developed by the Du Pont Engineering Department and were normalized to previous reactor cost studies for systems of this type.[4,5]

In the "Steam Plant" block of the code, the design of the turbine-generator plant, as well as the design of the steam generators for the liquid-D_2O-cooled reactors which employ indirect steam cycles, are optimized automatically. The annual charges associated with the steam generator construction, the inventory and pumping of D_2O, and the area of the building required to house this equipment are minimized. The steam that is delivered to the turbine is saturated. Thus, for a constant turbine exhaust pressure, the efficiency of the power conversion cycle can be expressed simply in terms of the steam pressure at the turbine throttle valve and the temperature of the feedwater from the turbine plant. The latter temperature is set by using the optimum number of regenerative

feedwater heaters in the cycle for each turbine throttle pressure. In the code, the throttle steam pressure is varied automatically until the steam plant, when combined with the previously designed reactor plant, gives the lowest total cost of electricity. The design and cost information required to derive the equations that make up this portion of the code as well as the information required to compute the cost of all the equipment and auxiliaries external to the reactor and its primary system, were supplied by Sargent & Lundy, Engineers.[6]

In the final block of the code, "Cost Accounting," the total capital costs including the D_2O inventory, the direct and indirect investment charges, the fuel cycle costs, and all the other operating charges are totalled to arrive at the final cost of energy for the system. Most of the analyses to date have been in accordance with the AEC ground rules (i.e., 14.3% per annum charge on plant investment, 12.5% per annum on D_2O at $28 per pound, outright ownership of the natural uranium fuel, AEC reprocessing of spent fuel, and $9.50 credit for each gram of recovered plutonium). These charge rates in the computer program can be varied in order to study the influence of different accounting procedures on the power reactor optimization.

The Cost Optimization Code was originally prepared for use with an IBM-650 digital computer equipped with six magnetic tape units. The time required for each problem was approximately ninety minutes. Although there was some automatic optimization of design parameters, most of the parameters had to be "manually" optimized by the elementary procedure of systematically comparing the results of individual problems. Approximately forty problems were required to optimize the major variables of a particular reactor concept within a given set of engineering and metallurgical limits and cost accounting procedures. More recently, the code has been translated for use with an IBM-704 computer which reduces the computing time per problem to about three minutes. In addition, the

greater capacity and flexibility of this machine permits more
optimizations to be carried out automatically. Accordingly, a
Latin Square technique is being developed for the computer to
optimize several of the design variables simultaneously.

The development of this code was a major undertaking as
evidenced by the complexity of the calculations required to establish
the design and cost of a power reactor plant. The preparation of
information in a form suitable for the code and the writing of the
code itself required the individual efforts of about two dozen men;
a total of about four man-years was required. However, this effort
is still only a relatively minor fraction of the total research and
development work being conducted on heavy water power reactors under
the auspices of the USAEC.

Use of Code

Many advantages accrue through use of computer codes of the
type described above. Within just a few minutes of computing
time, calculations are performed that would have required many
weeks if done by hand. The computer achieves a degree of internal
consistency seldom approached in cost evaluations done by hand.
In seeking an optimum design, many more design variations than
were practical heretofore can be evaluated with ease. New
technological and economic developments are readily incorporated
into the code making reoptimization a simple task. Thus, up-to-date
comparisons on a strictly consistent basis are always available
for each of the reactor concepts handled by the code. In the
paragraphs that follow we give some specific examples of the value
of the code in assisting to find answers to the management problems
posed in the introduction to this paper.

The first question concerned identification of the preferred
design concept for a heavy-water-moderated power reactor fueled
with natural uranium. A definitive answer to this question still
cannot be given. The research and development program on these
reactors is still underway. Even with the presently defined

engineering and materials criteria, full optimization of the design variables still has not been completed. Nonetheless, the results obtained to date in preliminary optimizations are presented in Table I in order to indicate the type of comparison we intend to establish upon completion of the study. Although the primary emphasis in this study has been on D_2O power reactors that use natural uranium fuel, the effect of designing these reactors with slight enrichment was also investigated by means of the cost optimization code; a uranium-oxide-fueled reactor containing the optimum enrichment is included in Table I for comparison. As can be seen from this table, all of the natural-uranium D_2O reactor concepts handled by the code have calculated total costs of energy that fall within a relatively narrow band. Thus, the results of current research and development efforts could significantly affect the relative economic merit of these reactor designs.

The second question posed in the introduction was whether the D_2O power reactors could compete economically with other types of power plants. Again it is too early to give a definitive answer to this question, but Table II illustrates the comparative costs among the natural-uranium D_2O reactor designs, a typical fossil-fueled station and an enriched-uranium boiling-H_2O reactor (i.e., the Pacific Gas and Electric - General Electric Plant, to be built at Bodega Bay, California).[7] All costs in Table II were normalized to the same AEC accounting rules as were used for the heavy water reactors. This table shows that the current designs of D_2O power reactors are not far enough advanced to produce energy as cheaply as the fossil-fueled or the boiling-H_2O plants in the United States. On the other hand, it must be recognized that the boiling-H_2O power reactor at Bodega Bay is the third large-scale plant of its type and the fossil-fueled plant represents an extremely well-developed technology. The estimated costs for the D_2O power reactor are based on a first-generation plant design.

Perhaps even more significanc to management than the second question then is the third; what is the future economic potential of the heavy-water power reactor concept? Table III lists some of the reductions in calculated energy costs that may be achieved by advances in the materials and engineering design limits currently used in the cost analysis. It must be emphasized that these are potential gains for advancements that may indeed prove to be beyond the technological capabilities of the system. However, it should also be pointed out that these listed gains do not include any savings in capital cost that would probably be made in later generation plants in which full advantage could be taken of lessons learned from operation of the first large-scale heavy water power reactors. Comparison of Tables II and III shows that although there is no significant economic advantage for the D_2O power reactors at the present time, the potential of these reactors appears sufficiently encouraging to warrant continuation of their study, especially in view of the inherent advantages associated with the use of natural uranium fuel.

The fourth question listed in the introduction was also intimately related to the economic potential of the D_2O power reactors; it concerned the economic climate that would favor these reactors over other types of electric-generating stations. The answers to this question are of great importance to management in their assessment of the potential market for these reactors, both at home and abroad. Table IV compares the calculated cost of energy from a D_2O power reactor, a fossil-fueled plant and a boiling H_2O power reactor for different capital charge rates. The particular reactors shown in this table have not been optimized for these new charge rates. They are the same reactors given in Table II but their energy costs have been recalculated for the new charge rates in order to demonstrate the effects of such changes on the relative economic merit of the systems. Heavy water power reactors are characterized by relatively high investment and very low fuel costs.

Thus, an economic environment with low capital charges puts greater emphasis on low operating costs and therefore favors the D_2O reactors. Such low charge rates do exist in foreign countries and in some places in the United States where the power system is federally or municipally financed. The potentially lower cost for D_2O noted in Table IV represents an estimated cost for heavy water produced in a new large-scale plant[8] and as indicated can cause a substantial decrease in the cost of energy from D_2O power reactors. The use of accounting procedures that involve outright ownership of fuel and a "throw-away" fuel cycle (i.e., the spent fuel is stored indefinitely rather than being chemically reprocessed to recover the manufactured plutonium and the remaining U-235) penalizes the H_2O boiling reactors much more than it does the D_2O power reactors by virtue of the latter's use of low-cost natural uranium fuel. The fuel cost in a fossil-fueled station of course depends on the location of the plant. The results of these calculations show that at present D_2O power reactors could compete favorably with other types of power plants in high fuel cost areas with low capital financing charges. In addition, in areas where outright purchase of the fuel and independence from reprocessing plants are desirable, the relative merit of the heavy-water natural-uranium power reactors increases in comparison with other nuclear power plants.

The fifth question was related to the research and development effort required to establish the feasibility of improvements that offer significant economic advantages over the current designs. It is in assisting to answer questions in this area that the code plays one of its most significant roles for management. The Cost Optimization Code can rapidly calculate the effects that a small change in one part of the system has on the rest of the system and on the total cost of energy. This ability makes the code an invaluable tool for assessing the potential worth of research and development efforts aimed at devising methods by which a calculated improvement could be achieved. For example, a 0.005-inch decrease in the thickness of the cladding on the metal fuel of a liquid-D_2O-

cooled power reactor results in a smaller temperature drop across
the cladding and in increased reactivity which in turn result in
changes in the optimum values for the fuel exposure, fuel tempera-
ture, coolant temperature, coolant pressure, pressure tube thickness,
power output per assembly, turbine pressure, power plant efficiency,
reactor size, steam generator size, turbine size, and over-all
plant size. Through use of the computer code, this decrease in
cladding thickness was calculated to result in an energy cost saving
of 0.16 mill/kwh. In a similar way, the cost reductions associated
with changes in other design variables and limits (e.g., increased
fuel temperature limit, increased heat flux limit, increased design
stress in the Zircaloy pressure tubes, etc.) were determined.
Several of these cost differentials are given in Table V. These
cost differentials place an immediate value on the research and
development work aimed at providing the means of obtaining the new
design value in practice. It is also evident from the preceding
discussion, that if many such small changes in the design variables
of a highly complicated plant or process are to be evaluated, a
computer program, such as the Cost Optimization Code for heavy-water
power reactors, is a practical necessity.

Conclusions

The Cost Optimization Code developed by Du Pont in cooperation
with Sargent & Lundy, Engineers, is proving to be an invaluable
tool in helping management to formulate the continuing program on
heavy-water power reactors and to answer the difficult questions
associated with the economic merit and the preferred course of
development for these reactors. The code is currently being expanded
to include additional design concepts (e.g., organic-cooled D_2O-
moderated reactors). If a D_2O power reactor were to be built, the
code would undoubtedly pay for itself many times over in direct
dollar savings by making it possible to design the best D_2O power
reactor for any given set of design criteria. The method of attack
used in this study of heavy water reactors has been very successful

and the high speed computing techniques used here appear to be quite amenable not only to other power reactor systems but to other management problems in other industries as well. Wherever there are difficult management problems arising from the design of highly complicated systems, consideration should be given to using such devices in aiding their solution.

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Table I

Comparison of Current Designs of 300-eMW D_2O Power Reactors

Fuel	Natural U Metal tubes	Natural UO_2 tubes	Natural UO_2 tubes	1.2% enriched UO_2 tubes
D_2O coolant	Liquid	Liquid	Boiling	Liquid
Maximum fuel temperature, $^{\circ}C$	500	2000	1000	2000
Maximum power per assembly, MW	8.54	5.64	3.96	8.77
Maximum heat flux, pcu/hr-ft^2	535,000	380,000	140,000	460,000
Cladding thickness, inch	0.020	0.020	0.020	0.020
Minimum coolant annulus thickness, inch	0.150	0.125	0.125	0.200
Number of fuel assemblies	175	244	305	125
Average fuel exposure, MWD/MT	5300	7100	7900	16,000
Over-all plant efficiency, %	25.1	24.0	27.2	27.4
Energy Costs, mill/kwh				
Plant	4.73	4.85	4.54	4.54
D_2O	0.85	1.05	1.62	0.73
Fuel	1.19	1.03	0.87	0.94
Operating	1.00	1.04	1.19	0.89
Total	7.77	7.97	8.22	7.10

Table II

Energy Costs from Various 300-eMW Power Plants

Fuel	Natural U	Natural UO_2	Natural UO_2	Enr. U[a]	Coal[b]
Moderator	D_2O	D_2O	D_2O	H_2O	-
Coolant	Liq. D_2O	Liq. D_2O	Boil. D_2O	Boil. D_2O	-
Energy Costs, mills/kwh					
Plant	4.73	4.85	4.54	3.9	3.3
D_2O	0.85	1.05	1.62	-	-
Fuel	1.19	1.03	0.87	2.26	3.3[c]
Operating	1.00	1.04	1.19	0.6	0.4
Total	7.77	7.97	8.22	6.76	7.0

Notes: (a) PG&E-GE Bodega Bay Plant, Reference 7

(b) Reference 3

(c) 35 cents/10^6 BTU coal

Table III

Effects of Technical Advances[a] on the Potential Cost of Energy from D_2O Power Reactors with Natural Uranium Fuel

1. **Natural Uranium Metal Tubes, Liquid D_2O Coolant**

   Maximum fuel temperature increased from 500 to 550°C

   Cladding thickness decreased from 0.020 to 0.015 inch

   Maximum fuel exposure increased from 8500 to 10,500 MWD/MT

   Design stress in pressure tube increased from 12,500 to 25,000 psi

   Resultant reduction in energy cost from 7.77 to 7.24 mills/kwh

2. **Natural Uranium Oxide Tubes, Liquid D_2O Coolant**

   Maximum fuel temperature increased from 2000 to 2500°C

   Cladding thickness decreased from 0.020 to 0.015 inch

   Design stress in pressure tube increased from 12,500 to 25,000 psi

   Resultant reduction in energy cost from 7.97 to 7.48 mills/kwh

3. **Natural Uranium Oxide Tubes, Boiling D_2O Coolant**

   Maximum heat flux increased from 140,000 to 300,000 pcu/hr-ft^2

   Cladding thickness decreased from 0.020 to 0.015 inch

   Design stress in pressure tube increased from 12,500 to 25,000 psi

   Resultant reduction in energy cost from 8.22 to 7.09 mills/kwh

Note: (a) The technical feasibility of operating at the advanced value of the design variable is not necessarily assured.

Table IV

Effect of Charge Rates on Power Reactor Merit

	Financing Charges, %/yr		Energy Costs, mills/kwh					
	Priv.[c]	Govt.[d]	D_2O-Nat. U.		Fossil Fuel		H_2O-Enr. U.	
			Priv.	Govt.	Priv.	Govt.	Priv.	Govt.
Plant	14.3	6.7	4.73	2.22	3.3	1.5	3.9	1.83
D_2O[a]	12.5	3.7	0.85	0.25	-		-	
Fuel[b]	-	-	1.19	1.19	3.3	3.3	2.26	2.26
Operating	-	-	1.00	1.00	0.4	0.4	0.6	0.6
Total			7.77	4.66	7.0	5.2	6.76	4.69

Notes: (a) Reduction of the cost of D_2O from \$28 to \$17 per pound
(Reference 8) would result in additional savings of
about 0.5 and 0.2 mill/kwh for privately financed and
government-financed plants, respectively.

(b) If outright ownership of the fuel and independence from
reprocessing are required, the natural-uranium heavy-water
power reactor is penalized about 0.2 mill/kwh; the enriched-
uranium light-water plant may be penalized more than
0.5 mill/kwh.

(c) Private (i.e. investor-owned) financing rates suggested in
AEC ground rules.

(d) Rates typical for government-financed plants.

Table V

Cost Differentials for 300-eMW D$_2$O Power Reactors Fueled with Natural Uranium

	Variation[a]		Potential Saving[b,c]	Reactor
	From	To	m/kwh	System
Maximum fuel temperature, $^{\circ}$C	500	550	0.12	U Metal
Maximum exposure, MWD/MT	8,500	10,000	0.17	tubes, liquid
Clad thickness, inch	0.020	0.015	0.16	D$_2$O coolant
Maximum fuel temperature, $^{\circ}$C	2,000	2,500	0.07	UO$_2$ tubes,
Clad thickness, inch	0.020	0.015	0.18	liquid D$_2$O
Minimum D$_2$O annulus, inch	0.150	0.125	0.10	coolant
Stress in pressure tube, psi	12,500	25,000	0.25	
Maximum heat flux, pcu/hr-ft^2	140,000	200,000	0.43	UO$_2$, boiling D$_2$O
Power distribution flattening	None	Optimum	0.14 to 1.4	U or UO$_2$, liquid or boiling D$_2$O

Notes: (a) The technical feasibility of operating at all the listed values of the design variable is not necessarily assured.

(b) The total cost of energy from these reactors is between about 7-1/2 and 8 mills/kwh.

(c) The indicated cost savings are not all additive.

FIGURE 1

SIMPLIFIED FLOW DIAGRAM OF THE COST OPTIMIZATION
CODE FOR HEAVY WATER POWER REACTORS

CODE BLOCK	MAJOR INPUT	MAJOR OUTPUT
Fuel Assembly Design	Pounds of uranium per foot, number of tubes per assembly, materials, lattice pitch, coolant flow area, inlet temperature and pressure.	Fuel assembly dimensions, maximum power output per assembly, physics parameters of infinite lattice, pressure drop across assembly.
Core Design	Fuel design, power flattening, required thermal power	Number of assemblies, reactor size, pumping requirements, fuel throughput.
Fuel Fabrication	Fuel design, annual fuel throughput	Fuel fabrication costs.
Reactor and Auxiliaries	Reactor core design	Design and costs of shields, pumps, pipes, minimum-cost containment vessel, control and safety equipment.
Steam Plant	Reactor coolant inlet and outlet temperatures, reactor power.	Design and costs of optimum steam generators and turbine plant for given reactor, net electrical output.
Cost Accounting	Annual charges on plant and D_2O investment, operating and maintenance costs.	Total investment, annual costs, fuel cycle costs, and TOTAL ENERGY COST IN MILLS PER KWH.