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OAK RIDGE, TENNESSEE

1 August 1958

Mr. J. B. Philipson
Operations Division
Atomic Energy Commission
Idaho Office of Operations
Idaho Falls, Idaho

Dear Mr. Philipson:

The enclosed memorandum ORNL CF 58-8-4 is an outline of the technical scope of the Gas-Cooled Reactor fuel element irradiation program as presently envisioned. This document will be followed by a second to be issued on 15 August which will delineate the actual reactor site assignments for the various experiments and the detailed cost estimates of the program.

The principal technical objectives of the irradiation program are the determination of the geometric stability and the fission gas release of the UO_2 fuel elements as influenced by temperature, density, heat flux and irradiation exposure. This program will also provide information on the thermal cycling properties of the UO_2 in the various fuel element configurations of interest, the relative merits of solid and annular fuel pellets and an evaluation of crushed oxide as a fuel element material. The overall aim is to provide a sufficient technical basis for selecting a final fuel element design by 1 December 1959.

The irradiation experiments, of which there are some 55 in all, are of three types. The first type (experiments 1 through 8) utilize the actual 0.705 in. OD prototype reactor fuel pellets and are intended in large part to serve as proof tests. The second type (experiments 9 through 42) utilize 0.5 in. OD pellets and involve a wide range of test conditions, geometries and materials. These experiments will help also to evaluate the pellets supplied by various vendors who may be considered as supplier for the first reactor core. The third type of experiment (numbers 43 through 55) is intended primarily for the purpose of studying the fission gas release.

Although final reactor site assignments for the various experiments will not be made until the final 15 August document is prepared, we have included an illustrative

DISCLAIMER

set in the present memorandum in an attempt to indicate feasibility and availability.

The program as presently outlined is in our opinion the minimum effort required for establishing an adequate fuel element design.

Sincerely yours,

Robert V. Meghreblian

Robert V. Meghreblian
Associate Director
Gas-Cooled Reactor Project

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Abstract

A set of 55 experiments has been outlined to provide a minimum irradiation program for selection of UO_2, pellet geometry and fabrication techniques, and canning technology. These experiments fall into three categories: prototype units in which radial dimension and heat fluxes are close to proposed design values, but irradiation times are long; reduced-size prototype for accelerated tests in which most variables will be studied; and miniaturized pellet irradiation to obtain high burnup for fission gas release studies. Reactor space has been found generally available and several installations are now examining their capabilities to participate in the program. A tentative schedule has been drawn to illustrate the feasibility of the program.

I. INTRODUCTION

In accordance with instructions from the Atomic Energy Commission, the Oak Ridge National Laboratory has initiated a program to provide fuel elements for the Prototype Gas-Cooled Reactor. For present scheduling purposes the dates appearing in ACF-GCPR-412, Revision 1, are considered firm.

This schedule requires a manufacturing program beginning approximately December 1, 1959. In advance of the manufacturing program a fuel rod* development and irradiation testing program is required to provide as much design data for firm specifications as is possible by December 1, 1959, and to provide data by June 1, 1959 for determining the behavior of UO_2 in a variety of geometric forms. The technical scope of the development program is described below.

Considerable ground work has been accomplished at NRTS, GE (Vallecitos, California), Curtiss-Wright (Quehanna, Pennsylvania), and ORNL toward implementing the development program and determining its cost. These will be described in a separate document to be issued August 15, 1958 following receipt and evaluation of proposals to accomplish the irradiations.

The fuel element development program will be the joint responsibility of the Reactor Projects, Metallurgy and Solid State Divisions of the Oak Ridge National Laboratory. The minimum program deemed necessary by the Laboratory comprises 55 test irradiations and will be augmented only as circumstances and future data indicate.

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* Various interpretations of terminology have been encountered. In this document (a) a fuel element assembly is a single cluster of fuel rods; (b) a fuel rod is any one of the several rods comprising a fuel element assembly; (c) a prototype test capsule is a portion of a proposed fuel rod having the radial dimensions proposed for a reactor fuel rod, but reduced in length for test purposes; (d) a reduced prototype test capsule is a reduced scale version of the prototype test capsule; and (e) a miniature test capsule is of an even smaller scale and designed to provide only selected specific data of interest.

Determination of properties of fuel rod cladding have been outlined in Kaiser-ACF Development Program 4.2 and procedures for fabrication and pre-irradiation inspection are being studied under 7.1, 7.2 and 7.3. Heat transfer and pressure drop characteristics are covered in 1.1 and 1.2. These aspects are not repeated below. The over-all combination of experiments is believed adequate to establish the reliability of proposed fuel element designs, and at present a pre-reactor irradiation test of a full-scale fuel element assembly does not appear to be warranted.

II. DEFINITION OF THE PROBLEM

The two main radiation problems of a stainless steel clad UO_2 fuel element are geometric stability and fission gas release. These are intimately related to temperature, density and heat flux.

Indications are that at moderate surface temperatures the fission gas release rate is not great, but that it is strongly dependent on temperature and density. Other factors which may affect the rate of fission gas release are method of manufacture, U/O ratio, UO_2 geometry, and rate of fuel burnup.

It is expected that the UO_2 will crack due to stresses caused by the large thermal gradients in the pellets. It is not known to what extent this will affect the integrity of the fuel rod when combined with the external pressure on the cladding. Some investigation has shown that once the pellet cracks to relieve the stress, further fragmentation on repeated cycling is not likely.

The effect of the release of the fission gases is to decrease the heat conductivity of the encapsulated gas and the transfer of heat across the gap between the UO_2 and the stainless steel, which in turn increases the temperature of the uranium oxide to cause a further increase on the rate of release of the fission gases. However, a point is reached where thermal radiation predominates so that the temperature of the fuel becomes insensitive to the gap conductivity or width. As part of the radioactive fission gases come out of the UO_2 they decay to stable gases which exert their partial pressure together with the initially encapsulated helium and result in a pressure buildup which might reach serious proportions over a long fuel life. One method of relieving this pressure is to use hollow cylinder (cored) fuel pellets which allows more gas volume in the fuel rod assembly. The hollow cylinder, however, provides voids into which UO_2 fragments can fall if serious UO_2 breakage or flaking should occur. Filler pieces or bushings of hollow or porous graphite or magnesium oxide within the cored pellets will be evaluated as a means of minimizing dislocation of UO_2 fragments. Going to porous UO_2 or powdered UO_2 would not be advantageous if this results in a higher fission gas release rate.

Pressure stresses on the rod encapsulation result from the external pressure of the helium coolant as well as the internal fission gas pressure. Thus, it is probable that a reversal of pressures will occur for high burnup elements or where reactor pressures are reduced for maintenance purposes. These as well as thermal stresses may be most important in the vicinity of welds and end pieces. Canning stability will be studied primarily under the 7.1 program, but will be included as a variable where possible in the irradiation program.

The irradiation test program must provide data for: (a) a choice of solid fuel pellets vs. annular configuration with or without filler pieces by an early date, July 1, 1959; (b) thermal cycling of capsules to simulate or exceed the cycling which would be experienced in the GCPR throughout the lifetime of a fuel charge. Since effects of thermal cycling are also pertinent to the choice of pellet geometry, significant data from early experiments will be required by June, 1959; (c) specification of manufacturing processes for UO_2 powder and pellets; (d) evaluation of crushed oxide as a fuel material; and (e) information from which the final element design can be specified; this to be completed by December 1, 1959.

It is not possible within the time afforded or the irradiation space available to more than provide data for simple choice of material, pellet geometry and manufacturing procedures. No provision has been made for a development program in the event that serious defects in the design are discovered in the course of the tests outlined. This event, although unlikely, would require revision of the program, and extended schedule, and probably additional test equipment.

III. PARAMETRIC CONSIDERATIONS

The irradiation program will study the properties of UO_2 in 304 stainless steel test capsules under conditions at least as severe as those anticipated in the reactor. Cladding temperatures will be 1300°F or higher, and power level (except for the miniature capsules) will be 30,000 to 50,000 Btu/hr/ft. Burnup will vary from 3,000 to 13,000 MwD/MT depending on the nature of the variables under study. The experimental program will provide data on the inter-relationships of the following variables:

A. Bulk UO_2:

1. Fission gas release as a function of:
   a. density
   b. temperature
   c. enrichment
   d. burnup rate
   e. burnup
   f. thermal stress
   g. manufacturing process
   h. uranium-oxygen ratio
   i. geometric form
2. Stability of the UO_2 body as a function of:
   a. variables as listed in A-1 above
   b. stresses other than thermally induced
   c. thermal cycling
   d. UO_2 geometry; i.e., comparison of solid pellets, cored pellets, cored pellets with internal bushing, and powdered or crushed UO_2
3. Pre- and post-irradiation observations:
   a. density
   b. thermal properties
      (1) thermal conductivity
      (2) specific heat
      (3) stored energy

c. dimensions

d. strength tests

e. Young's Modulus

f. X-ray diffraction examination

g. petrographic examination

h. metallographic examination

i. visual examination

j. pressure buildup due to fission product gases

B. Individual Capsule:

1. Strength of the encapsulating geometry

   a. to withstand attack by or leakage of internal or external gases

   b. to withstand distortion due to external or internal pressures

2. Weld integrity

3. Condition of UO_2 body

   a. extent of fragmentation

   b. effects of fragmentation on deformation mechanisms in the capsule

4. Effects of thermal cycling on deformation mechanisms

5. Effects of corrosion

6. Correlation of manufacturing processes and inspection techniques and determination of techniques adequate to meet the required reactor service conditions

7. Comparison of the general integrity of capsules enclosing either:

   a. cored UO_2 pellets

   b. cored UO_2 pellets with internal graphite or MgO bushings

   c. solid UO_2 pellets

   d. powdered or crushed UO_2

IV. EXPERIMENTAL CONDITIONS

A listing of experiments including the principal variables considered has been prepared and is shown in Table I. The experiments A, B and C are currently in progress, having been initiated for the GCR-2 program. As can be noted, the heat flux for the prototype capsule (experiment B) is lower than that required for the Kaiser-ACF design.

The design cladding temperature has been specified for most tests and it is proposed that this be maintained as closely as is experimentally possible. Several higher temperature experiments are included to evaluate possible hot spot temperature conditions. Heat fluxes have been chosen to give an approximation of the thermal stress at the surface of the pellet. Central temperature for hollow pellets are held below 3000°F and are less than 4000°F for solid UO_2 cylinders. Burnup and the test periods are the longest considered possible within the time limits for the program.

The experiments are grouped according to pellet size and the corresponding test conditions. The first group, Numbers 1 through 8, are of full scale radial dimension for both pellets and cladding. The power levels were chosen to bracket the probable design value of 40,000 Btu/ft/hr for the Kaiser-ACF fuel rod. Four of the test periods obviously extend beyond the program completion date, but have been included to match the fuel rod design conditions. Obviously, if information from these tests is deemed desirable at an earlier date, they can be terminated at that time.

The second group of experiments, reduced prototype, embracing Numbers 9 through 42 includes most of the variables to be examined. These include pellets of different geometries, and powdered or granular UO_2, thermal cycling and steady state experiments, oxide densities lower than the specified 95%, higher cladding temperature and varying burnup. Pellets supplied by different vendors who may be considered as supplier for the first core would be evaluated in this set of experiments. These experiments are designed around smaller pellets to accelerate the test period without increasing the heat fluxes and thermal stress beyond reasonable limits.

The miniature experiments of the third group, Numbers 43 through 55, evaluate fission gas release as a function of UO_2 density, oxygen ratio, temperature and burnup.

This is considered to be a minimum program to afford a basis for selection of materials and geometry for the first core fuel loading. Little direct duplication has been included for statistical analysis although some basis for statistical treatment is afforded by repetition of certain variables in experiments where other parameters are under study. In all cases where thermal cycling is to be programmed, duplicate test pieces are to be provided concurrently in the same reactor so that effects of cycling from reactor shutdown and startup can be differentiated from those in the program.

Table I GCFR Capsule Irradiation Program (Cont't)

Experiment Number		Size of UO_2 (in.) OD	ID	Actual Time of Irradiation Wk	Da	Burn-Up Total $\frac{MwD}{MT}$	Rate $\frac{MW}{MT}$	Fission Power Btu/hr/ft	Fission Heat Flux Btu/hr ft^2	Flux Av Perturbed	Density % Theory	Temp Cap. Sur. °F	Heat Transfer Media	Enrichment
Miniature Prototype														
46		.156	.078	11	79	7,000	89.0	7,000	172,000	2.5×10^{13}	75	1500	Gas	10.0
47		.156	.078	13	89	7,000	78.4	7,000	172,000	2.0×10^{13}	85	1500	Gas	10.0
48		.156	.078	13	89	7,000	78.4	7,000	172,000	2.0×10^{13}	85	1500	Gas	10.0
49		.156	.078	27	186	13,000	70.0	7,000	172,000	2.0×10^{13}	95	1500	Gas	10.0
50		.156	.078	27	186	13,000	70.0	7,000	172,000	2.0×10^{13}	95	1500	Gas	10.0
51	V	.156	.078	14	100	7,000	70.0	7,000	172,000	2.0×10^{13}	95	1500	Gas	10.0
52	V	.156	.078	14	100	7,000	70.0	7,000	172,000	2.0×10^{13}	95	1500	Gas	10.0
53	V	.156	.078	14	100	7,000	70.0	7,000	172,000	2.0×10^{13}	95	1500	Gas	10.0
54		.156	.078	14	100	7,000	70.0	7,000	172,000	2.0×10^{13}	95	1300	Gas	10.0
55		.156	.078	14	100	7,000	70.0	7,000	172,000	2.0×10^{13}	95	1700	Gas	10.0

B - Bushing in central hole, TC - Thermal cycling, - Pairs control test with TC test,
V - To evaluate pellets supplied by potential vendors.

Table I GCPR Capsule Irradiation Program

Experiment Number	Size of UO_2 (in.) OD	Size of UO_2 (in.) ID	Actual Time of Irradiation Wk	Actual Time of Irradiation Da	Burn-Up Total $\frac{MwD}{MT}$	Burn-Up Rate $\frac{Mw}{MT}$	Fission Power Btu/hr ft	Fission Heat Flux Btu/hr ft^2	Flux Av Perturbed	Density % Theory	Temp Cap. Sur. °F	Heat Transfer Media	Enrichment
GCR-2													
A	.156	.078	10	72	6,000	83	7,000	118,000	2.0×10^{13}	95	1500	Gas	10.0
B	.156	.078	10	72	6,000	83	7,000	118,000	2.0×10^{13}	95	1500	Gas	10.0
C	.705	.25	27	189	1,500	80	21,000	105,000	1.0×10^{13}	95	1300	Gas	1.8
I Prototype													
1	.705	.323	128	894	13,000	14.6	31,400	161,000	2.0×10^{13}	95	1300	Gas	2.0
2	.705	.323	128	894	13,000	14.6	31,400	161,000	2.0×10^{13}	95	1300	NaK	2.0
3	.705	.323	85	596	13,000	21.9	47,200	241,000	2.0×10^{13}	95	1300	Gas	3.0
4	.705	.323	29	206	3,000	14.6	31,400	161,000	2.0×10^{13}	95	1300	Gas	2.0
5	.705	.323	20	137	3,000	21.9	47,200	241,000	2.0×10^{13}	95	1300	Gas	3.0
6	.705	0	26	183	3,000	16.4	44,700	228,000	1.5×10^{13}	95	1300	Gas	3.0
7	.705	0	26	183	3,000	16.4	44,700	228,000	1.5×10^{13}	95	1300	NaK	3.0
8	.705	.323	20	137	3,000	21.9	47,200	241,000	2.0×10^{13}	95	1300	NaK	3.0
II Reduced Prototype													
9	.5	.25	48	334	13,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
10	.5	.25	26	180	7,000	38.9	40,000	280,000	1.0×10^{14}	95	1500	Gas	1.0
11	.5	.25	26	180	7,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
12	.5	0	34	240	7,000	29.2	40,000	280,000	8.0×10^{13}	95	1300	Gas	1.0
13	.5	0	34	240	7,000	29.2	40,000	280,000	8.0×10^{13}	95	1300	NaK	1.0
14	.5	.25	26	180	7,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	NaK	1.0
15] B	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
16] BTC	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
17]	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
18] TC	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
19]	.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1500	Gas	1.0
20] TC	.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1500	Gas	1.0
21]	.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1300	Gas	1.0
22] TC	.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1300	Gas	1.0

B - Bushing in central hole, TC - Thermal cycling,] - Pairs control test with TC test,

V - To evaluate pellets supplied by potential vendors.

Table I GCPR Capsule Irradiation Program (Cont'd)

Experiment Number		Size of UO_2 (in.) OD	ID	Actual Time of Irradiation Wk	Da	Burn-Up Total MWD/MT	Rate MW/MT	Fission Power Btu/hr ft^2	Fission Heat Flux Btu/hr ft^2	Flux AV Perturbed	Density % Theory	Capsule Temp Cap. Sur. °F	Heat Transfer Media	Enrichment
Reduced Prototype														
23	B	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
24	BTC	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
25		.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
26	TC	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
27		.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1500	Gas	1.0
28	TC	.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1500	Gas	1.0
29		.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1300	Gas	1.0
30	TC	.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1300	Gas	1.0
31	V	.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1300	Gas	1.0
32	V	.5	0	15	103	3,000	29.2	40,000	280,000	8.0×10^{13}	95	1300	Gas	1.0
33	V	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
34	V	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
35	V	.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
36		.5	.25	11	77	3,000	38.9	40,000	280,000	1.0×10^{14}	95	1300	Gas	1.0
37		.5	.25	9	65	3,000	46.2	40,000	280,000	1.5×10^{14}	80	1300	Gas	1.0
38		.5	.25	10	73	3,000	41.0	40,000	280,000	1.0×10^{14}	90	1300	Gas	1.0
39		Crushed		8	54	3,000	55.6	40,000	280,000	1.5×10^{14}	50/95 BC	1300	Gas	10.0
40		Crushed		8	54	3,000	55.6	40,000	280,000	1.5×10^{14}	50/95 BC	1300	NaK	10.0
41		Crushed		11	76	3,000	39.6	40,000	280,000	1.0×10^{14}	70/95 BC	1300	NaK	10.0
42		Crushed		11	76	7,000	39.6	40,000	280,000	1.0×10^{14}	70/95 BC	1300	NaK	10.0
III Miniature Prototype														
43		.156	.078	27	186	13,000	70.0	7,000	280,000	2.0×10^{13}	95	1500	Gas	10.0
44		.156	.078	27	186	13,000	70.0	7,000	280,000	2.0×10^{13}	95	1500	Gas	10.0
45		.156	.078	11	79	7,000	89.0	7,000	172,000	2.5×10^{13}	75	1500	Gas	10.0

B - Bushing in central hole, TC - Thermal cycling, - Pairs control test with TC test,

V - To evaluate pellets supplied by potential vendors.

V. METHOD OF APPROACH

In order to accomplish the extensive irradiation program within the specified periods, it is necessary to limit experiment design to the simplest assemblage which can produce meaningful results. Devices are chosen to provide a limited number of variables each with multiplication of experiments to produce the total program. Since temperature is a highly important variable, experiments are limited to those facilities in which instrument lines can be accommodated, and thermocouples will be attached in practically all cases.

Simple capsules may be used with a very thin gas annulus surrounding the pellet can to dissipate heat directly to test reactor cooling water. Since the high heat flux limits the annulus to a few mils for the 1300°F can temperature, tolerance variations may produce hot spots exceeding those for the GCPR elements. An alternate experiment design utilizes pressurized NaK as the capsule environment with an outer gas annulus which can be machined to close tolerances as a heat barrier. Since NaK is a good conductor of heat, hot spot effects should virtually be eliminated and the test would be less severe than the design requirement. Thus, the two experiments bracket the desired conditions. Each would be pressurized to 400 psia to match the reactor coolant pressure.

More elaborate tests involve the use of once through air systems or recirculating helium loops. These facilities can match the design requirements, but for reasons of economy and expediency may be limited to lower pressures and lower gas temperatures than for the GCPR. Generally, these loops are more costly than a capsule program despite their capability of utilizing simpler test pieces, but appear justifiable for a portion of the program to obtain prototype test conditions. Accordingly, the suitability of several facilities are being considered and proposals are to be made for test equipment of this type.

Thermal cycling may be accomplished by providing a cyclic moving neutron curtain for one or more of the gas loops or capsules. Space limitations restrict this arrangement to a very few facilities. Existing equipment may be available at one installation to move capsules in and out of the reactor core.

In order to obtain fission gas release information rapidly, a series of very small pellets will be irradiated to high burnup at different oxide densities and temperatures. This test rig has already been used for two tests in a previous program and can be duplicated readily. Unfortunately, the pellet size is so small that oxide mechanical stability cannot be evaluated.

Prime consideration will be given to the suitability of each reactor for the various types of experiments and to the utilization of existing equipment where possible. The various tests will be placed to utilize the different reactor fluxes to best advantages and the number of pellet configurations and uranium enrichments will be limited to minimize procurement problems. Accurate determination of reactor fluxes present a difficult problem for some facilities. It is proposed to install the simplest capsule in some of these facilities as soon as possible to obtain data on flux and flux depression as well as early irradiation information and samples with which to gain experience in hot cell examination.

VI. IRRADIATION FACILITIES UNDER CONSIDERATION

Discussions have been held with reactor engineering, operations and hot cell personnel at the MTR and ETR, NRTS; at the GETR, Vallecitos, California; BMI, Columbus, Ohio; and CWRR, Quehanna, Pennsylvania. Representatives of ORNL, IDO, Kaiser Engineers and ACF Industries participated in conversations concerning irradiation facilities. Discussions with BMI were held in Oak Ridge with ORNL personnel only. Discussions pertaining to hot cell facilities and capabilities were conducted by ORNL with representatives of each of the sites indicated. A list of experiments essentially as shown in Table I was discussed with each group and their capabilities to conduct these experiments were reviewed. Each group contacted agreed to consider the program, to determine its interest in participation and, if interested, to submit a proposal in time for consideration for the August 15 program submission.

Briefly, irradiation space has been indicated to be available as shown in Table II.

Table II

Reactor	No. of Holes	Apprx Thermal Flux	Notes
LITR	4-6	2×10^{13} neut/cm^2/sec	Small loops or capsules
ORR	4-10	10^{14}	Capsule irradiation Thermal cycles
MTR	2-3	10^{14}	Capsule irradiation
ETR	10-20	10^{14}	Capsule irradiation
GETR	18	10^{14}	Loops or capsule in pool Thermal cycles
CWRR	6-10	2×10^{13}	Loops or capsule in pool reflector
BMI	6-10	$<10^{13}$	In core

The LITR appears attractive for once-through air-cooled (low flow) loops to cool the group of miniature capsule experiments, Numbers 44 through 55 of Table I. It has convenient access to the tank and an off gas system for handling moderate quantities of air. Tank access to the ORR is more difficult, limiting its usefulness to static capsule irradiations for the required time schedule. Access to the ORR pool, however, is relatively simple and is suitable for most of the irradiations.

Access to both the MTR and ETR is comparable to that for the ORR and again these reactors are considered for capsule irradiations. Very little space is available in the MTR and its general crowded conditions make it unattractive for lead-type experiments. The cyclers provided by KAPL may be available for thermal cycling experiments in the MTR, but these have not been used successfully with lead-type experiments and their usefulness in this program is questionable. The ETR offers the greatest amount of available space in a reactor which is now operating and should be useful for an extensive capsule program.

The GETR with its large pool area at high flux (1×10^{14} neut/cm^2/sec) offers the most convenient access and greatest potential versatility for this experimental program. A number of irradiation spaces appear to be available and suitable. Thermal cycling experiments can be designed with rotating or oscillating shutters which are directly operable by virtue of the pool access. Also convenient access is provided for loop type experiments and short piping runs are afforded by the proximity of equipment rooms. The GETR is scheduled for operation in December, 1958 and this date appears reasonably firm.

The CWRR is now equipped for a 1 Mw power level and is just starting operation. Assurance has been given of 3 Mw operation by January, 1959, which would provide useable fluxes for this program. It has been proposed that beryllium pieces would be fabricated to form a reflector immediately surrounding the core and to contain from 6 to 10 experimental irradiation spaces suitable for this program. Once-through air cooling could also be accommodated.

Flux levels in the BMI reactor with the possible exception of one central core position are too low for consideration in this program. (Also BMI has declined to submit a proposal for this program.)

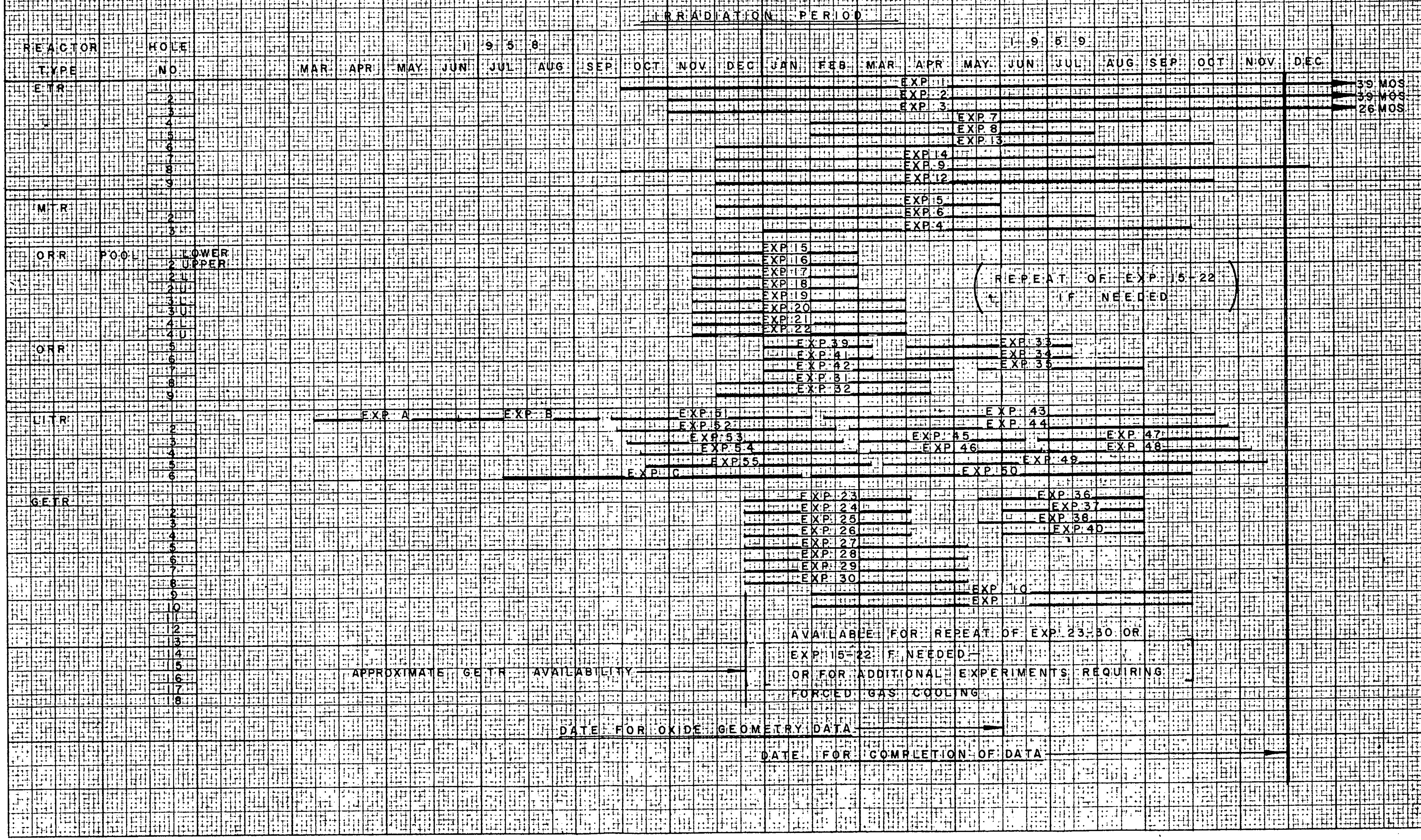
VII. SCHEDULE CONSIDERATIONS

A preliminary schedule is shown as Table III. It is designed to accommodate the experiments described in Table I in types of facilities as described in Section VI and to meet the requirements of the two basic deadlines, June 1 and December 1, 1959. Each experiment has been fitted to the reactor type believed to be best able to supply the needed technical information. The assignments indicated are illustrative, although complete flexibility is not possible within the specified deadlines and available facilities. Actual assignments will be outlined in the final report to be released August 15.

Consideration has been given to basic details of supplying oxides, cladding materials, shop and inspection facilities and the feed rates appear feasible although the entire schedule is extremely tight. Time has also been allowed following each irradiation for hot cell examinations and, provided the irradiation schedule is adhered to, will not exceed the estimated capacity of seven examinations per month.

The experiments basically needed for data on UO_2 geometry are shown as concurrent irradiation experiments through 22. As confirmatory tests for oxide geometry evaluation, experiments 23 through 30 are provided at a second reactor site. Those to evaluate powdered oxide as irradiation experiments 39 and 42, and those to evaluate oxides from different vendors as irradiation experiments 31 through 35. Experiments to evaluate fission gas release in irradiated capsules are shown as experiments 43 through 55. Experiments 1 to 14 and 36 to 38 are a group in which different variables are under study.

TABLE III PRELIMINARY SCHEDULE



VIII. POST-IRRADIATION EXAMINATIONS

Concurrently with negotiations for irradiations, investigations have been made to ascertain the availability of hot cells for post-irradiation observations. Oak Ridge National Laboratory will have one cell available by January 1, 1959; General Electric Vallecitos Laboratory has four cells which appear to be the likely choice for the bulk of the work; Curtis-Wright has hot cell capacity which also may be used.

The basic plan is to negotiate CPFF contracts for hot cell work in the amount needed, to ship irradiated capsules to these facilities and to conduct all examinations under the direct supervision of Oak ~~Ridge~~ National Laboratory personnel. Evaluations and criteria derived from these examinations will then be made available for improvement of design, fabrication and inspection procedures preparatory to manufacturing the reactor fuel rods.

The following measurements will be made in all experiments:

(1) Macroscopic examinations and inspection of the fuel pellet container for blistering, bulging or other distortion and macroscopic inspection of the UO_2 for evidence of fusion, cracking or chipping.

(2) Dimensional measurements of the outside of the stainless steel can and any of the UO_2 pellets which remain whole.

(3) Photographs of any unusual conditions such as weld failure or bulging of the cans.

(4) Macroscopic examination including metallographic examination of the UO_2 for changes in grain size or structure.

(5) Metallographic examination of the container and hardness test on the container.

(6) Density of the UO_2.

(7) Burnup analysis of the UO_2.

(8) Fission gas analysis and pressure buildup.

(9) Analysis for fission product deposition on inside wall of the stainless steel can.

VIII. POST-IRRADIATION EXAMINATIONS (continued)

The following properties are of long-term interest and will be performed whenever equipment is available for doing the tests:

(1) Stored energy

(2) Thermal conductivity

(3) Specific heat

(4) X-ray defraction

(5) Petrographic examination

(6) Elastic modulus

(7) Strength

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