UNCLASSIFIED

ORNL-2256
Engineering

**AQUEOUS HOMOGENEOUS RESEARCH
REACTOR - FEASIBILITY STUDY**

P. R. Kasten
M. I. Lundin
C. L. Segaser

**OAK RIDGE NATIONAL LABORATORY
OPERATED BY**

UNION CARBIDE NUCLEAR COMPANY

A Division of Union Carbide and Carbon Corporation

POST OFFICE BOX X • OAK RIDGE, TENNESSEE

UNCLASSIFIED

DISCLAIMER

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

DISCLAIMER

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

35

Printed in USA. Price _____ cents. Available from the

Office of Technical Services
U. S. Department of Commerce
Washington 25, D. C.

— LEGAL NOTICE —

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

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

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

UNCLASSIFIED

ORNL-2256

I

Contract No. W-7405-eng-26

REACTOR EXPERIMENTAL ENGINEERING DIVISION

AQUEOUS HOMOGENEOUS RESEARCH REACTOR - FEASIBILITY STUDY

P. R. Kasten	M. I. Lundin	C. L. Segaser
R. E. Aven	H. A. McLain	
D. R. Gilfillan	R. A. McNees	
R. F. Hughes	C. Michelson	
M. C. Lawrence	C. W. Nestor	

DATE ISSUED

MAY 9 1957

OAK RIDGE NATIONAL LABORATORY
Operated by
UNION CARBIDE NUCLEAR COMPANY
A Division of Union Carbide and Carbon Corporation
Post Office Box X
Oak Ridge, Tennessee

UNCLASSIFIED

UNCLASSIFIED

II

INTERNAL DISTRIBUTION

ORNL-2256
Engineering
TID-4500 (12th ed.)

1. C. E. Center
2. Biology Library
3. Health Physics Library
4. Metallurgy Library
- 5-6. Reactor Experimental
Engineering Library
- 7-9. Central Research Library
- 10-14. Laboratory Records Department
15. Laboratory Records, ORNL R.C.
16. G. M. Adamson
17. J. E. Baker
18. S. E. Beall
19. A. M. Billings
20. D. S. Billington
21. E. P. Blizard
22. E. G. Bohlmann
23. C. J. Borkowski
24. G. E. Boyd
25. J. C. Bresee
26. F. R. Bruce
27. W. D. Burch
28. C. A. Burchsted
29. A. D. Callihan
30. D. W. Cardwell
31. W. L. Carter
32. G. H. Cartledge
33. R. A. Charpie
34. R. D. Cheverton
35. H. C. Claiborne
36. W. E. Clark
37. T. E. Cole
38. E. L. Compere
39. D. D. Cowen
40. S. Cromer
41. F. L. Culler
42. J. S. Culver
43. J. S. Drury
44. W. K. Eister
45. L. B. Emlet (K-25)
46. D. E. Ferguson
47. J. H. Frye, Jr.
48. C. H. Gabbard
49. J. L. Gabbard
50. W. R. Gall
51. H. E. Goeller
52. A. T. Gresky
53. J. C. Griess
54. E. Guth
55. P. H. Harley
56. P. N. Haubenreich
57. J. W. Hill
58. C. J. Hochanadel
59. A. Hollaender
60. A. S. Householder
61. G. H. Jenks
62. W. H. Jordan
63. S. I. Kaplan
64. P. R. Kasten
65. C. P. Keim
66. M. T. Kelley
67. R. B. Korsmeyer
68. A. S. Kitzes
69. K. A. Kraus
70. J. A. Lane
71. W. J. Leonard
72. R. E. Leuze
73. S. C. Lind
74. R. B. Lindauer
75. R. S. Livingston
76. M. I. Lundin
77. R. N. Lyon
78. J. L. Marsh (C&CCC, South Charleston)
79. H. F. McDuffie
80. J. P. McBride
81. R. A. McNees
82. J. R. McWherter
83. O. Menis
84. E. C. Miller
85. K. Z. Morgan
86. E. J. Murphy
87. M. L. Nelson
88. E. O. Nurmi
89. J. P. Murray (Y-12)
90. L. F. Parsly
91. Sigfred Peterson
92. M. L. Picklesimer
93. L. R. Quarles
94. W. D. Reel
95. P. M. Reyling
96. A. M. Rom
97. M. W. Rosenthal
98. W. L. Ross

UNCLASSIFIED

UNCLASSIFIED

99. H. C. Savage	113. D. S. Toomb
100. C. H. Secoy	114. W. E. Unger
101. C. L. Segaser	115. R. Van Winkle
102. E. D. Shipley	116. B. Weaver
103. M. J. Skinner	117. A. M. Weinberg
104. R. W. Stoughton	118. E. P. Wigner (consultant)
105. A. H. Snell	119. G. C. Williams
106. I. Spiewak	120. J. C. Wilson
107. C. D. Susano	121. C. E. Winters
108. J. A. Swartout	122. F. C. Zapp
109. E. H. Taylor	123. ORNL – Y-12 Technical Library,
Document Reference Section
110. D. G. Thomas	124. HRP Director's Office, Y-12
111. P. F. Thomason	
112. M. Tobias	

EXTERNAL DISTRIBUTION

- 125. Houdry Corporation (Attn: G. A. Mills)
- 126. Division of Research and Development, AEC, ORO
- 127-133. TISE for P.A.R.P.
- 134-709. Given distribution as shown in TID-4500 (12th ed.) under
Engineering category (200 copies – OTS)

UNCLASSIFIED

IV

UNCLASSIFIED

CONTENTS

SUMMARY	1
1. TECHNOLOGICAL FEASIBILITY	3
1.1 Fuel-Solution Composition	3
1.2 Oxygen Addition	3
1.3 Materials Problems	3
1.4 HRR Materials Recommendation	3
1.5 Temperature Effects	5
1.6 Velocity Effects	5
1.7 Stress Corrosion	5
1.8 Radiation Corrosion	5
1.9 Effects of Irradiation on Strength of Metals	6
2. NUCLEAR FEASIBILITY	7
2.1 Critical Calculations	7
2.2 Flux Depression In and Around Thimble	8
2.3 Nuclear Considerations in Control and Startup	9
2.4 Reactor Safety	10
2.5 Buildup of Fuel Isotopes and Fission-Product Poisons	10
2.6 Heat Production in Thimbles and Pressure-Vessel Wall	11
2.7 Biological-Shielding Requirements	11
2.8 Converter Design	11
3. PROCESS OPERATIONS	13
3.1 Flowsheet	13
3.2 Internal Recombination of Radiolytic Gases	13
3.3 Fission-Gas Disposal	13
3.4 Startup and Shutdown	13
3.5 Critical Temperature	16
3.6 Thimble-Rupture Hazard	17
3.7 Process Design	17
3.8 Normal Operation	17
4. EXPERIMENTAL FACILITIES	19
4.1 Types of Experiments	19
4.2 Description of Facilities	19
4.3 Thimble Design	19
4.4 Design Proposals	21
5. MECHANICAL DESIGN OF REACTOR COMPONENTS	23
5.1 High-Pressure System	23
5.2 Low-Pressure System	26
5.3 Conceptual Layouts of Reactor Complex	28
6. PROCESSING OF FUEL SOLUTION	35
6.1 Flow-Diagram Description	35
6.2 Preliminary Equipment Arrangement	35

UNCLASSIFIED

UNCLASSIFIED

7. ECONOMIC EVALUATION	38
7.1 Turbogenerator Station	38
7.2 Cost Estimate	38
ACKNOWLEDGMENTS	42
BIBLIOGRAPHY	42

AQUEOUS HOMOGENEOUS RESEARCH REACTOR - FEASIBILITY STUDY

SUMMARY

The results of a preliminary investigation into the feasibility of an aqueous homogeneous research reactor (HRR) for producing a thermal flux of 5×10^{15} neutrons/cm².sec are presented herein. This study indicates the present-day technological feasibility of the proposed reactor and defines the nature of the design and development problems associated with the construction of the reactor complex. It is concluded that design could proceed on the basis of the assumed parameters with confidence that successful operation could be achieved, although some problems will require development study. The reactor studied would operate at 500 Mw of heat and generate 125 Mw of electric power.

The essential objective of this report is to describe the problems associated with an aqueous homogeneous reactor that would provide high thermal- and fast-neutron fluxes for experimental use. Beyond this objective, the designs are carried only to the stage of completion required to make a preliminary layout.

This study shows that, although no technological breakthroughs are necessary, major problems do exist in the categories of maintenance (philosophies and detailed design planning), operational reliability (design and development of large-scale components), and experimental facilities (thimbles, corrosion, cooling, mechanical joints, accessibility, and conversion of thermal to fast flux).

Because of the flexibility of aqueous homogeneous reactors with regard to changes in fuel concentrations and removal of fission- and corrosion-product poisons, their use as high-flux reactors (in which high fuel burnup is a primary consideration) appears attractive. Certain restrictions have to be met, however, which involve such interrelated quantities as total reactor power, gas generation rate, corrosion, neutron-flux requirements, and the limitations of existing technology. The number and complexity of the requirements which enter into the analysis required an early and somewhat arbitrary selection of the reference design.

The study was based on a single-region reactor with enriched (8%) uranium, in the form of uranyl

sulfate (10 g of uranium per kilogram of D₂O), as the fuel and D₂O as the moderator and coolant.

Solution temperature limits were set at 225 to 275°C to avoid the two-phase separation phenomena at higher temperatures and to minimize the corrosion difficulties. Sufficient copper sulfate is added to the solution to recombine 100% of the radiolytic gases produced. This helps maintain solution stability from a safety consideration, reduces the required system overpressure, and simplifies the reactor complex by minimizing the low-pressure-system requirements. Total system pressure is set at 1400 psia. Excess sulfuric acid is required to stabilize the copper sulfate, uranyl sulfate, and corrosion-product nickel. Based on corrosion-limited wall power densities, a 10-ft-dia stainless-steel-lined carbon-steel sphere was chosen, which results in a total power requirement of approximately 500 Mw for a maximum thermal-neutron flux of 6.5×10^{15} neutrons/cm².sec at 250°C. Under these conditions the approximate power densities would be 2 kw/liter at the core wall, 35 kw/liter average, and 110 kw/liter maximum. For experimental purposes a maximum thermal flux of about 3×10^{15} neutrons/cm².sec and a virgin-neutron flux of about 5×10^{14} neutrons/cm².sec would be available.

To minimize corrosion of equipment and piping in the external circuit, all flow velocities are held to values below the critical velocities. In the reactor vessel, a corrosion rate of 70 to 80 mpy is predicted for the Zircaloy-2 experimental thimbles and 10 mpy is predicted for the stainless steel liner of the reactor vessel, provided that the flow velocity is not allowed to exceed 3 fps.

Fission- and corrosion-product solids will be produced at a rate of approximately 20 lb per day under normal reactor operating conditions. These solids will be concentrated into 750 liters of fuel solution per day by means of hydroclones with self-contained underflow pots. The rare-earth sulfates will be dissolved, and a suitable wash cycle will separate the major portion of the uranium from the bulk of the fission- and corrosion-product solids. The uranium may be returned to the reactor system. The chemical processing

cycle for recovery of uranium and D_2O will also tend to limit fission and corrosion products to tolerable amounts in the reactor system. The corrosion-product nickel must be limited to 0.01 M if two-liquid-phase equilibrium is to be maintained between 225 and 275°C.

One of the principal advantages of an aqueous homogeneous type of reactor is its inherently simple control system. It has been calculated that the HRR would have a temperature coefficient of reactivity at 250°C of approximately $-2.5 \times 10^{-3}/^{\circ}C$ and at 20°C of approximately $-9 \times 10^{-4}/^{\circ}C$, which, in combination with fuel-concentration control, would permit safe operation without control rods.

Extensive consideration must be given to the design and layout of the external systems and of the experimental facilities, since the heat generated in the reactor core will be removed by pumping the solution through an external heat-exchanger system. For maximum accessibility to the experimental facilities, the heat-exchanger equipment will be placed in separate shields in compartments at a distance from the reactor core, resulting in large inventories of D_2O and fuel.

All maintenance must be done by remote methods, as all equipment will be highly radioactive owing to contamination with fuel solution. The philosophy adopted for the HRR, as reflected in the layouts shown, is based on the premise that a dry maintenance procedure can be developed by utilizing mobile and stationary shielding together with remotely operated and specially designed tooling.

Experimental facilities are provided in seven 6-in.-ID solid-wall Zircaloy-2 thimbles. The severity of the problem of Zircaloy-2 corrosion must be determined experimentally. Another problem in the thimble design lies in developing the method of remotely joining Zircaloy-2 to stainless steel. An apparently satisfactory joint was proposed for the HRT system, but because of the higher pressure differential in the HRR (1400 psi as compared to ~ 400 psi in the HRT), development of a suitable joint will be required.

The over-all utility of the reactor would be enhanced if, in addition to a high thermal flux, a high fast (>1 Mev) flux were attainable. This can be accomplished either by a converter plate or by a concentrated fuel solution exposed to the thermal flux. Since power densities would be in the range of hundreds of kilowatts per liter, the burnout problem would limit exposures if a converter plate were used. A fuel-solution type of converter, with

new fuel solutions being added continuously, would permit long exposures. This type, with its extensive fuel handling and auxiliary systems, would be more complex than the plate type.

Calculations for a fluid-fuel type of converter based upon cylindrical geometry indicate that a virgin flux of about 4×10^{14} neutrons/cm²·sec can be attained at the center of a 1-in. hollow cylinder having a power density of 500 kw/liter. The total heat generated would amount to approximately 1 to 2 Mw. Calculations based upon spherical geometry are expected to reveal that fluxes higher than 4×10^{14} will be available.

Radiations from the reactor vessel will be reduced to biological and instrument tolerance by 9 ft of high-density concrete — 6 ft around the reactor proper and 3 ft at the containment-vessel periphery. Six feet of shielding is specified around the heat exchanger and low-pressure-equipment cells.

With the proposed design a sudden rupture or a major leak in the high-pressure-equipment containment vessels can result in a 50-psi equilibrium expansion pressure.

Normally, operating personnel will be excluded from the hot cells adjacent to the experimental thimbles during reactor operation because of the hazard associated with a ruptured thimble. In this event, a large portion of the fuel solution could escape to the hot cell and containment vessel. This contingency could be reduced by resorting to a thimble designed to have an attached pressure-balancing chamber, which would be expected to retain the solution if the thimble should rupture. If the fuel solution were not retained by the chamber, it could not in any event escape from the containment vessel; this containment vessel will be designed for the equilibrium expansion pressures which might occur. With the reactor shut down, there would be sufficient biological shielding to permit limited access to the hot cell containing the experimental facility.

Although the production of economical power was not considered a major design objective, the dissipation of the heat energy by means of the generation of electrical power will provide valuable information and experience in the design of future power producers based on the aqueous homogeneous reactor principle. The installation of a turbogenerator station would pay for itself if the power produced could be sold for 2 to $2\frac{1}{2}$ mills/kwhr.

1. TECHNOLOGICAL FEASIBILITY

Research on some of the major feasibility problems anticipated in the design and development of large-scale circulating-fuel homogeneous reactors has been carried on during the past few years at ORNL. In particular, much emphasis has been placed on the problems of fluid-fuel types and compatible materials of construction. The basic design parameters for the HRR are listed in Table 1, and it is believed that design could proceed with confidence that successful operation could be achieved at the desired temperatures.

1.1 FUEL-SOLUTION COMPOSITION

Based on the technology and fundamental knowledge now available, a uranyl sulfate solution is an acceptable fuel. Nuclear considerations require the use of D_2O as the solvent if total power is to be held at a minimum while the flux is held at the desired level. Total recombination of the radiolytic gases at a core outlet temperature of $275^\circ C$ requires a copper concentration of 0.02 moles/liter, and a total system pressure of 1400 psi (see Fig. 6). The amount of excess sulfuric acid required to stabilize the copper sulfate, uranyl sulfate, and corrosion-product nickel against hydrolytic precipitation has been set at 0.02 moles/liter.

An initial fuel enrichment of 8% will be necessary for criticality with a fuel concentration of 10 g/liter at an operating temperature of $250^\circ C$ and 10% poisons. With fully enriched uranium and with the above temperature and poison conditions, a U^{235} concentration of only 0.645 g/liter would be required. Because fuel solutions containing less than 10 g of uranium per liter have not been adequately studied, the enrichment was decreased to stay in the realms of existing technology.

Uranyl sulfate solutions containing 10 g of uranium per liter and enough excess acid to prevent hydrolytic precipitation of the uranium and copper can be satisfactorily contained in stainless steel at temperatures ranging from 225 to $275^\circ C$. The recommended fuel composition and the uranium enrichment are summarized in Table 2.

1.2 OXYGEN ADDITION

Corrosion of metals by uranyl sulfate solutions results in reduction of the uranyl ion by precipita-

tion of the uranium as UO_2 and formation of free sulfuric acid unless an oxidizing agent is added to maintain the oxidation state of the uranium. Dissolved oxygen has proved to be an effective oxidant. Knowing the corrosion rate (representing the yearly loss of metal thickness) and the exposed area, the amount of oxygen combining with metal may be calculated as approximately 7.8 g/year-mil-ft².

1.3 MATERIALS PROBLEMS

A major problem in the design and construction of a practical large-scale homogeneous reactor is that of determining suitable materials for the piping and equipment. The requirements of such materials are:

1. They must be resistant to corrosion by uranyl sulfate solutions over the operating range of temperature, concentrations, velocities, and irradiation.
2. They must have sufficient strength at elevated temperatures to safely contain the solutions at high pressure.
3. Their properties should not be changed appreciably by long-time irradiation from the reactor.
4. In some cases, as in the materials for experimental thimbles, it is mandatory that the material have a low capture cross section for thermal neutrons.

These requirements drastically curtail the materials available for use in the HRR system.

1.4 HRR MATERIALS RECOMMENDATIONS

Based on the limitations imposed by strength, corrosion, and radiation damage, the following materials of construction were chosen for the HRR:

1. reactor vessel – ASTM A-212-B carbon-steel base material clad with type 347 stainless steel; HRP specifications;
2. experimental thimbles – hafnium-free Zircaloy-2; HRP specifications;
3. piping – type 347 stainless steel; HRP specifications;
4. circulating pumps – type 347 stainless steel base material, with titanium as an alternate impeller material.

TABLE 1. KEY DESIGN PARAMETERS

Reactor type	Single-region, circulating-fuel, homogeneous
Fuel type	$\text{UO}_2\text{SO}_4\text{-D}_2\text{O} + \text{CuSO}_4 + \text{H}_2\text{SO}_4$
Amount of U^{235}	45.8 kg
Uranium concentration	
Total uranium	10 g/liter at 250°C
U^{235}	0.8 g/liter at 250°C
CuSO_4 to recombine 100% of gas	0.02 M
H_2SO_4 to stabilize uranium and copper	0.02 M
Maximum nickel concentration	0.01 M
Fuel-solution temperature	
Minimum (inlet to reactor vessel)	225°C
Maximum (outlet of reactor vessel)	275°C
Average (system)	250°C
Fuel-system pressure	1400 psi
Neutron flux (experimental)	
Maximum thermal	$3\text{--}4 \times 10^{15}$ neutrons/cm ² ·sec
Maximum fast in 1-in.-dia cylindrical converter	4×10^{14} to 1×10^{15} neutrons/cm ² ·sec
Power density	
Maximum (at reactor center)	110 kw/liter
Average	34 kw/liter
Minimum (at thermal shield)	2 kw/liter
Total heat generated	500 Mw
Reactor-vessel key specifications	
Inside diameter	10 ft
Vessel material	Carbon steel clad with type 347 stainless steel
Total volume	14,800 liters
Net fluid volume (approximate)	12,000 liters
Experimental facilities	
Horizontal	6
Vertical	1
Maximum inside diameter	6 in.
Material	Zircaloy-2
Minimum wall thickness	$\frac{3}{4}$ in.
Maximum wall thickness	1 in.
External system	
Material	Type 347 stainless steel, HRP specifications
Fluid volume (external system only)	34,000 liters
Allowable velocities	
225°C	10–15 fps
250°C	25–35 fps
275°C	30–40 fps
Reactor control	Negative temperature coefficient of reactivity; changes in concentration of fuel
Heat dissipation	Generation of approximately 125 Mw of electrical power

TABLE 2. FUEL COMPOSITION AND URANIUM ENRICHMENT

Fuel solution	$\text{UO}_2\text{SO}_4\text{-D}_2\text{O} + \text{CuSO}_4 + \text{H}_2\text{SO}_4$
Uranium concentration	10 g/liter at 250°C (0.042 M)
U^{235} concentration	0.8 g/liter (at 250°C)
CuSO_4 to recombine 100% of gas	0.02 M
H_2SO_4 for stabilizing solution	0.02 M

1.5 TEMPERATURE EFFECTS

From the standpoint of safety in the mechanical design of reactor components, especially the experimental thimbles, a low fuel-solution temperature is desirable because of the associated low vapor pressure. However, since the efficiency of copper as an internal catalyst decreases rapidly with the lower solution temperatures, the operating temperature must be high enough for essentially 100% recombination of the radiolytic gases. On the other hand, there is a temperature above which a solution of any given uranium concentration will separate into two phases. Therefore the final design temperature was a compromise temperature which results in a system pressure low enough that experimental thimbles may be designed with reasonably thin walls, high enough for efficient internal recombination of gases, but not so high as to result in temperatures anywhere in the system at which phase separation would occur. Based on these premises, an entrance temperature of 225°C and an exit temperature of 275°C were selected, resulting in a 50°C rise in temperature through the reactor, with an average solution temperature of 250°C.

1.6 VELOCITY EFFECTS

From the results of recent corrosion loop tests, the critical velocities listed in Table 3 are recommended for the piping systems of the HRR (assuming type 347 stainless steel as the material of construction). Fuel velocities equal to or less than these values result in out-of-pile corrosion rates of approximately 1 mpy.

1.7 STRESS CORROSION

Although stress-corrosion cracking of type 347 stainless steel, titanium, or Zircaloy-2 in contact

TABLE 3. CRITICAL FUEL VELOCITIES IN THE PIPING SYSTEM

Temperature (°C)	Critical Velocity (fps)
225	10-15
250	25-35
275	30-40
300	40-50

with uranyl sulfate solutions has not been observed, the seriousness of the consequences of failure demands that freedom from susceptibility to this type of attack be completely and adequately demonstrated. Type 347 stainless steel specimens have been stressed to 40,000 psi for 1,560 hr while being exposed to 0.04 M solutions of UO_2SO_4 with 0.006 M H_2SO_4 and 0.005 M CuSO_4 at 300°C without cracking or unusual attack. However, there is evidence to support the belief that the presence of fluorides, iodides, or other halogens may accelerate stress corrosion. Information as to maximum permissible concentrations is not available.

1.8 RADIATION CORROSION

In-pile loop experiments have shown that the effect of radiation on the corrosion of zirconium alloys and type 347 stainless steel is appreciable. The corrosion rate of Zircaloy-2 under reactor irradiation in contact with uranyl sulfate solutions (0.042 M) appears to be a function of power density as follows:

$$C = \alpha(P)^{0.6},$$

where

C = corrosion rate, mpy,

P = power density, kw/liter,

α = constant which is related to temperature as follows:

Temperature (°C)	α
225	3.5
250	1.7
280	1

A corrosion rate of 70 to 80 mpy has been estimated for the Zircaloy-2 experimental thimbles under the proposed HRR conditions.

In the reactor vessel the effect of irradiation is to reduce the permissible fuel velocity from approximately 30 fps at an average temperature of 250°C to 3 fps, with a resultant corrosion rate of about 10 mpy.

1.9 EFFECTS OF IRRADIATION ON STRENGTH OF METALS

Since experimental thimbles of Zircaloy-2 capable of withstanding 1400 psi external pressure are to be inserted into the high-flux region of the reactor, it is important to know the ultimate effects of neutron irradiation on the strength of the Zircaloy-2. Also, the shell of the reactor pressure vessel will be subjected to relatively high neutron fluxes, although not so high as those to which the thimbles will be exposed.

The equipment in future large-scale nuclear reactor power plants may be exposed to fast-neutron irradiations totaling as high as 2×10^{25} *nvt*. Specimens of carbon steel and austenitic stainless steel have been exposed in existing

reactors to maximum fast integrated fluxes of about 10^{20} *nvt*. This amount of exposure resulted in definite mechanical-property changes. In carbon steels the general effects of irradiation appear to be an increase in the yield stress, elimination or modification of the yield point, reduction of elongation and reduction of area, and an elevation of the ductile-to-brittle failure transition temperature in the Charpy impact test. The energy requirements for ductile failure are lowered. Carbon steel (A-212-B) irradiated in the MTR at 10^{20} *nvt* exhibited distinctly brittle behavior up to about 122°F. Below the transition temperature the energy for fracture is markedly less, and even above the transition temperature the energy for fracture is only about one-half that required to break unirradiated control specimens. Superficially, irradiation is capable of making austenitic stainless steel act like a pearlitic, low-carbon steel, with this important exception — the stainless steel appears to retain high ductility over a much wider temperature range, even at low temperatures.

2. NUCLEAR FEASIBILITY

This section contains the results of nuclear calculations for the HRR and covers such studies as critical calculations for a variety of reactor sizes, temperatures, and poison levels; power densities and total reactor power; flux depression in thimbles; safety calculations; production of fuel isotopes and fission-product poisons and their buildup with time; startup and shutdown procedures; heat-generation studies; and converter design. These results are complete only insofar as deemed necessary for a nuclear-feasibility evaluation of the reactor system.

2.1 CRITICAL CALCULATIONS

These calculations were of a survey type to evaluate the core-wall power density and total reactor power as a function of operating conditions for a fixed maximum thermal flux. Initially the geometry was assumed to be spherical, but calculations were also performed for cylindrical reactors, which indicated that spherical geometry is preferred on the basis of minimizing total power.

Fuel concentrations were calculated by using a simple bare-reactor model. The flux in the neighborhood of a boundary predicted by such a model is known to be somewhat higher than the actual flux, but, since the reactor is surrounded by a thick steel pressure vessel, some neutron reflection occurs, which tends to increase the flux at the boundary. As a first approximation, then, the values for wall flux and power density were computed from the simple model.

It would be desirable to minimize the total reactor power for a specific flux level. However, the associated core power density would result in undesirably high metal-corrosion rates. Therefore a compromise must be made between power level, thermal flux, and corrosion rates. A large reactor size would be particularly desirable with respect to corrosion rates at the experimental-thimble walls, since, for a fixed total power, the power density varies inversely with the reactor volume. There does not appear to be a difficult radiation-corrosion problem associated with the pressure-vessel wall itself, so long as a thick thermal shield is present. Corrosion rates of the thermal shield, however, would increase with decreasing size, because of the increasing wall power density. A 10-ft-dia reactor was selected as one which would give the desired average thermal flux with

acceptable values for the core and wall power densities.

To aid in selecting reactor size and operating conditions, critical mass, fuel concentration, power, maximum power density, and wall power density (at the thermal shield) were obtained for bare spherical reactors having diameters of 8, 10, 11, and 12 ft, operating at 20, 100, 200, and 300°C, and having poison levels of 0, 10, and 30% (based on macroscopic fission cross section). Maximum flux in all reactors was taken as 1×10^{16} neutrons/cm²·sec; since the power and all power densities are directly proportional to this figure, power and power densities corresponding to other fluxes can be easily computed. The values for power density at the thermal-shield wall are rough estimates only, since the flux at the wall is a sensitive function of the assumed extrapolation distance. Results for the conditions specified above are shown in Figs. 1 through 4.

Fig. 1. Critical Mass of Spherical One-Region Reactors as a Function of Temperature, Poison Level, and Reactor Diameter.

Fig. 2. Power Level Required to Obtain a Thermal Flux of 10^{16} neutrons·cm⁻²·sec⁻¹ in Spherical One-Region Reactors.

Fig. 3. Maximum Power Density Associated with a Thermal Flux of 10^{16} neutrons·cm⁻²·sec⁻¹ as a Function of Diameter in Spherical One-Region Reactors.

Fig. 4. Power Density at Thermal-Shield Wall as a Function of Diameter in Single-Region Spherical Reactors Having a Maximum Thermal Flux of 10^{16} neutrons·cm⁻²·sec⁻¹.

2.2 FLUX DEPRESSION IN AND AROUND THIMBLE

The thermal flux actually attainable within an experimental thimble is a sensitive function of the contents of the thimble. A detailed calculation was made for a 6-in.-ID, 1-in.-thick, cylindrical Zircaloy-2 thimble, of infinite length, filled with H₂O and situated at the center of a cylindrical reactor of infinite length. The reactor diameter required for criticality and the flux distribution were calculated, assuming the same average power

density and fuel concentration as those in a 10-ft spherical reactor. To estimate the effect of the thimble, the maximum flux in the cylinder with the thimble removed was obtained by means of a two-group, three-region calculation; the results are listed in Table 4.

The decrease of flux in the central region was due primarily to the change in flux distribution and the restriction of constant total power. This same effect was predominant in slow- to fast-flux converter calculations, in which the fast virgin flux inside a 3-in.-dia converter (cylindrical geometry) containing 50 g of U^{235} per liter was only $\sim 5 \times 10^{14}$ neutrons/cm²·sec (the central thermal flux was depressed from 6×10^{15} to $\sim 4 \times 10^{14}$ neutrons/cm²·sec by insertion of the converter, for constant total power).

2.3 NUCLEAR CONSIDERATIONS IN CONTROL AND STARTUP

In the temperature range of 225 to 250°C, the temperature coefficient of reactivity for the HRR would be about $-2 \times 10^{-3} \Delta k_e / ^\circ\text{C}$, and so inherent control appears to be adequate under normal power operation. At 20°C, $\partial k_e / \partial T$ would be controlled by the nuclear temperature coefficient, which would be about $-8 \times 10^{-4} \Delta k_e / ^\circ\text{C}$.

In addition, at 20°C there would be a density coefficient of reactivity of about $-1.7 \times 10^{-4} \Delta k_e / ^\circ\text{C}$ associated with fluid temperature rise. The total temperature coefficient would thus be about $-9 \times 10^{-4} \Delta k_e / ^\circ\text{C}$ at 20°C, which appears to be adequate to permit startup of the reactor at low temperatures. This implies that rates of reactivity addition associated with reasonable fuel-concentration rates would not be excessive. Increasing the fuel concentration at $\sim 20^\circ\text{C}$ would add reactivity at the rate of $0.5 \Delta k_e$ per doubling of fuel concentration. A permissible rate of reac-

tivity addition would be $\sim 0.002 \Delta k_e / \text{sec}$ (see Sec. 2.4), corresponding to a core pressure rise of ~ 250 psi and a very rapid temperature rise of 10 to 15°C during the initial surge. The fuel concentration could therefore be doubled in ~ 5 min from a nuclear-safety viewpoint. This would correspond to bringing the reactor from critical at 20°C to critical at $\sim 250^\circ\text{C}$, and so thermal aspects would be controlling rather than nuclear ones.

In the initial startup, it is suggested that the reactor be fueled with solution dilute enough to ensure subcritical conditions. Fuel solution could then be added at a rate consistent with the above results and thermal restrictions. If sufficient heating capacity (~ 5 Mw) is available for heating up fuel solution, it may be preferable to heat the fuel solution as it is added to the high-pressure system and to attain criticality with the reactor initially hot. This would eliminate gas-evolution difficulties if Cu^{++} catalyst were present; no nuclear difficulty appears to exist for this situation.

After power operation, there would be large amounts of after-heat energy due to fission-product decay. It may then be possible to shut the reactor down by partially dumping, letting the fission products keep the fluid at a high temperature, and circulating the fluid at a rate sufficient to keep the temperature from exceeding some limit. About half the fuel solution would need to be dumped, and the remainder could be maintained near operating temperature. Calculations indicate that following two weeks of operation and two days of shutdown, the fission fragments in the HRR would still generate ~ 1 Mw of heat energy. This would correspond to $\sim 1^\circ\text{C}$ of fluid temperature rise per minute if this heat were generated and retained within the core fluid volume. Thus, even though the fuel solution were at 100°C and the external

TABLE 4. FLUX DEPRESSION CAUSED BY THIMBLE

Critical radius with thimble inserted	125 cm
Critical concentration with thimble, 250°C, no poisons	0.46 g/liter
Critical concentration with no thimble, 250°C, no poisons	0.42 g/liter
Maximum thermal flux with no thimble, average power density = 34 kw/liter	6.15×10^{15} neutrons/cm ² ·sec
Thermal flux at thimble wall	3.95×10^{15} neutrons/cm ² ·sec
Thermal flux at center of thimble (H ₂ O filled)	3.87×10^{15} neutrons/cm ² ·sec

volume were twice the core volume, after-heat energy could heat the fluid to 250°C in ~7 hr. Under these conditions the problem associated with distillation of large quantities of fuel solution or heating the fuel solution by external means during startup could be alleviated.

2.4 REACTOR SAFETY

Because of the relatively high values for the temperature coefficient of reactivity ($\partial k_e/\partial T = -9 \times 10^{-4}$ to $-2.5 \times 10^{-3}/^\circ\text{C}$ over the temperature range 20 to 250°C) and the low value for prompt-neutron lifetime ($>10^{-3}$ sec), the HRR has a high degree of inherent safety.

The maximum core pressure rise, p_{max} , associated with a reactivity addition is a function of the reactivity addition, the reactor design, the temperature coefficient, prompt-neutron lifetime, and operating conditions. In this evaluation of reactor safety, the cross-sectional area of the core exit piping was taken as 3 ft², the length of piping between the core and pressurizer was assumed to be 10 ft, the pressurizer volume was 50 ft³, the core volume was 500 ft³, and the operating conditions were 250°C and 1500 psia. If enough reactivity were added to the reactor to make it 1% above prompt critical, the resulting core pressure rise would be about 250 psi.

The rate of reactivity addition equivalent to an instantaneous reactivity addition would be dependent upon the initial reactor power, P_0 . Assuming $P_0 = 0.04$ w (source power), a rate of reactivity addition of $0.002 \Delta k_e/\text{sec}$ would be equivalent to an instantaneous reactivity addition of 1% above prompt critical. Since this would result in a maximum core pressure rise of only 250 psi, $0.002 \Delta k_e/\text{sec}$ appears to be a conservative value with regard to permissible rates of reactivity addition.

Although the above results would be roughly applicable at either 250 or 20°C, the fluid temperature rise associated with these two base temperatures would vary considerably owing to the change in $\partial k_e/\partial T$ and would be greater at the lower base temperature.

2.5 BUILDUP OF FUEL ISOTOPES AND FISSION-PRODUCT POISONS

If the reactor were to operate with no fuel processing, the fuel isotopes and fission-product poisons accumulating within the reactor system would

tend to lower the neutron flux for a fixed operating power level. To indicate the magnitude of such an effect, the results obtained¹ for a 9-ft D₂O-moderated burner reactor operating at 250°C were applied to the HRR. The absolute numbers do not apply to the HRR (since highly enriched fuel was considered), but the relative results do, and these results indicate the effect of fission-product poisons on critical concentrations. The change in fuel concentration with time is given in Table 5, based on an operating power level of 500 Mw.

TABLE 5. BUILDUP OF FUEL CONCENTRATION WITH TIME, ASSUMING NO FUEL PROCESSING

Operating Time at 500 Mw (years)	Concentration (g/liter)			Fuel Enrichment* (%)
	U ²³⁵	U ²³⁶	U ²³⁸	
0	0.53	0	0.053	90
1	0.82			
2	0.92	0.72	3.4	18.4
5	1.03	1.76	7.9	10.7

*90% enriched feed assumed during period of operation.

The results in Table 5 indicate that the fission-product poisons would tend to decrease the flux by a factor of ~1.6 in one year and a factor of ~2 in five years. Thus, if any real advantage is to be gained by fission-product removal, these neutron poisons must be removed on a short-time scale (months). The preceding conclusions do not consider removal by hydroclones, which, if used, would appreciably decrease the above factors. If no poisons were removed, the required U²³⁵ concentration would change only slowly with time after two years of operation.

Another point to consider is the increase in total uranium content with time, due to the accumulation of U²³⁶ and U²³⁸. However, if highly enriched feed is used, it does not appear that there will be any detrimental accumulation.

At 500-Mw operation, ~1 lb of fission products would be formed per day of operation. If these accumulate as solids, it would be desirable to remove them from the reactor by hydroclones to

¹P. R. Kasten and H. C. Claiborne, *Nucleonics* 14(11), 88 (1956).

prevent the possibility of reactivity addition upon stoppage of fluid circulation. The fuel feed rate required would be ~ 600 g of U^{235} per day.

If all decomposition gases were recombined in solutions, then xenon would not be removed and would result in about 0.05 fraction poisons. This is not appreciable during operation, but after shut-down the xenon buildup could cause operational difficulty unless removed from the reactor.

2.6 HEAT PRODUCTION IN THIMBLES AND PRESSURE-VESSEL WALL

(a) **Experimental Thimbles.** — A preliminary calculation of the expected heat generation was made on the assumption that all gamma rays produced in the core which strike the thimble surfaces are absorbed. The number of gammas striking the thimbles was estimated from multiplying the total gamma production rate times the ratio of the surface area of the thimbles to the total surface area of the core and thimbles. Thus an average heat-generation rate due to the core gamma rays was obtained for both the fission and fission-product decay gammas. The production rate of thermal-neutron-capture gamma rays originating within the thimble wall was obtained on the basis of the average thermal flux; no attenuation of the thermal flux in the Zircaloy-2 thimbles was considered. Exponential attenuation with no buildup was assumed to hold for gamma rays passing through the thimble walls. For a 6-in.-ID, 1-in.-thick thimble, the heat generation from all gamma sources was 3.7 w/g, of which the neutron-capture gamma rays contributed about 2.6 w/g and the fission and decay gammas about 0.55 w/g each.

(b) **Pressure-Vessel Wall.** — Gamma-ray heating in the pressure-vessel wall would be due largely to the capture of thermal neutrons in the iron and subsequent production of gamma rays. The heating for a given thermal-neutron current into the wall was calculated by a method devised by Enlund and outlined by Chapman,² while the thermal-neutron current was calculated from reactor power on the basis of a bare-reactor approximation. This procedure yielded a heat-generation rate of 10.2 w/cc at the inner wall of a 6-in.-thick unshielded pressure vessel; the heat-generation rate decreased approximately exponentially after passing through

a maximum of 10.5 w/cc about 1 cm from the shell surface, to a value of 0.5 w/cc at 12 cm into the shell, and then gradually leveled off. The above results would be applicable to a thermal shield if it were placed within the core region to protect the pressure vessel from thermal stress.

2.7 BIOLOGICAL-SHIELDING REQUIREMENTS

Preliminary estimates indicate that about 9 ft of special concrete, containing 1% boron and a water content of 7% and having a density of 3.0 g/cc, would be required for shielding the reactor tank. This thickness could consist of two or more parts, one around the reactor core and the other surrounding associated equipment, compartmentalized to provide limited accessibility for equipment maintenance. The thickness of shield required for the high-pressure system external to the core region would be ~ 6 ft.

2.8 CONVERTER DESIGN

The problem of obtaining a high virgin flux is directly related to that of obtaining a high power density within the experimental facility. For a homogeneous solution a thermal flux of $\sim 10^{15}$ neutrons/cm²·sec would be required to obtain a virgin flux of 10^{15} neutrons/cm²·sec, and the U^{235} concentration would be in the approximate range of 10 to 50 g of U^{235} per liter. The power generated in this facility would be in the range of 1 to 5 Mw. The design difficulties would include those associated with containment of the solution, necessity for replacement, heat-removal problems (heat exchanger), and accessibility to a high-flux region.

Some of the above difficulties can be reduced by using a heterogeneous converter-plate design. However, long exposures (periods of several weeks) at high flux are not possible without converter replacement owing to the burnout problem.

A converter suitable for use inside a 3-in.-dia thimble of a homogeneous research reactor has been proposed for obtaining high fast-neutron fluxes. The limitation on the amount of fast-neutron flux that can be obtained from such a converter is dependent on the accessibility of a thermal flux of 10^{15} neutrons/cm²·sec and on the rate at which the generated heat can be removed from the converter.

A converter unit consisting of a series of thin concentric cylinders shows promise of producing fast-neutron fluxes of the order of $\sim 2 \times 10^{15}$

²R. H. Chapman, *Analysis of Spherical Pressure Vessel Having an Energy Source Within the Wall*, ORNL CF-54-10-76 (Dec. 26, 1954).

neutrons/cm².sec if a thermal flux of 10^{15} is available. This fast flux refers to the number of unmoderated neutrons passing through a 1-cm² area normal to the neutron velocity at the geometrical center of the cylinders. The proposed converter consists of seven concentric cylinders 4 in. long, 1 mm thick, with 3-mm spacing between cylinders. The radius of the inner cylinder is 0.48 in. and that of the outer cylinder is 1.42 in. Larger cylinder lengths do not appear desirable because of the attenuation of the fast neutrons by the cooling water. Thinner cylinder plates would permit higher neutron fluxes, and efforts should be made to reduce the thickness of these plates to the practical minimum if this type converter is to be

used. The power generated by such a converter would be ~2 Mw.

As indicated in Sec. 2.2, the presence of a converter has an appreciable effect upon the flux distribution, to the extent that the maximum thermal flux is drastically reduced for a specified total power. Therefore it does not appear possible to attain a thermal flux of 10^{15} within the converter volume, for the assumed total reactor power, the specified reactor size, and the converter geometry envisioned. It would be possible to attain the desired fast flux if insertion of the converter caused no appreciable change in spatial distribution of power density.

3. PROCESS OPERATIONS

3.1 FLOWSHEET

A schematic flowsheet for the HRR system is presented in Fig. 5. The 500 Mw of thermal energy produced by the nuclear reaction will be removed in six double-drum 83.3-Mw heat exchangers. Two 17,650-gpm pumps mounted on the outlet pipes of the heat exchanger circulate the reactor solution around the primary circuit. Saturated steam at 300 psia is generated at a rate of 2.11×10^6 lb/hr and may, subsequently, be used to generate 133,000 kw of gross electrical power at a cycle efficiency of 26.5%. A net power generation of 125,000 kw will be delivered at the station bus-bars, approximately 6% being required for station auxiliaries. Stainless steel piping in the primary fuel-solution circuit is sized to be compatible with the critical velocities listed in Table 3. The system shown is pressurized to 1400 psia by means of an externally mounted high-pressure steam generator. Because the technology of vapor pressurization is more advanced, this mode was selected for the HRR system in preference to gas pressurization, although either system could probably be made to perform satisfactorily. Feed water, consisting of D_2O from the condensate tank, is supplied to the steam generator through an economizer by means of a 0.5-gpm feedwater pump. This flowsheet does not contain a letdown system for separating and recombining radiolytic gases, since 100% internal recombination will be achieved at the system operating temperature of 250°C by means of internal copper catalyst. With the exception of the fission-gas disposal system, the remainder of the low-pressure dump-tank system has been determined on the basis of the assumed normal shutdown and startup procedures.

3.2 INTERNAL RECOMBINATION OF RADIOLYTIC GASES

Fissioning of uranium in uranyl sulfate solutions decomposes water into hydrogen and oxygen. The value of G_{D_2} is approximately 1.7 molecules of D_2 per 100 ev of energy absorbed for solutions of UO_2SO_4 in D_2O containing 10 g of uranium per liter. Cupric sulfate dissolved in the fuel solution has been found to recombine the hydrogen and oxygen as rapidly as they are formed, so that there is no net gas production in the reactor solution.

A preliminary study³ has been made to estimate the amount of $CuSO_4$ catalyst required in the HRR to internally recombine essentially 100% of the radiolytic gases produced. In the studies on the internal recombination in the HRT, an accurate analysis was made, taking into consideration the temperature variation in the core vessel and in the heat exchanger. It was found that if the average temperature was used, along with the saturation value for dissolved D_2 at the core exit, a reasonably accurate value would be obtained for the minimum copper concentration required for 100% internal recombination.

With no recombination the gas production rate in the HRR is calculated to be 1.75×10^6 cc of D_2 at STP per second or 0.172 lb mole of D_2 per second. If the fuel solution is assumed to be essentially all D_2O , then the number of pound moles of solution is equal to 3490, and the average deuterium production rate in the HRR is then 4.93×10^{-5} moles of D_2 per mole of solution per second.

Based on these premises the minimum copper requirements for 100% and for 80% internal recombination are given in Table 6 for three operating temperature and pressure ranges in the HRR. From this table a system pressure of 1400 psia and a solution containing 0.02 M $CuSO_4$ are recommended. Figure 6 is a plot of the total pressure required as a function of fuel temperature for various copper concentrations.

It should be noted that it may be profitable to consider the use of the silver ion as an internal catalyst for the recombination process. The investigation of silver appears desirable, since its lower activation energy may make operation of the HRR practical at pressures as low as 400 to 500 psia and at temperatures in the range of 150 to 250°C.

3.3 FISSION-GAS DISPOSAL

The method proposed for fission-gas disposal in the HRR is the same as that used in the HRE and HRT. The fission gases and excess oxygen are taken directly from the cold traps into a bed of

³R. E. Aven, *Internal Recombination in the HRR*, ORNL CF-56-4-190 (April 26, 1956).

Fig. 5. HRR Flowsheet.

TABLE 6. COPPER AND DEUTERIUM CONCENTRATION IN THE HRR

Temperature Range (°C)	Core Exit Temperature (°C)	p_T (psia)	p_{D_2} (psi)	K_{D_2} (mole fraction/psi)	y_{D_2} (sat)	Cu^{++} Concentration (M)	
						100% Recombination	80% Recombination
200–250	250	1000	280	3.40×10^5	0.00083	0.11	0.09
225–275	275	1440	385	2.63×10^5	0.00146	0.019	0.016
250–300	300	2000	500	2.20×10^5	0.00227	0.0043	0.0035

Fig. 6. Total Pressure Required vs Fuel Temperature in HRR Core Exit for Various Copper Concentrations.

activated charcoal. A steady-flow type of operation is used, with the bed providing sufficient holdup time to allow most of the radioactive isotopes to decay before the stream emerges from the bed. The gas leaving the charcoal bed is diluted and discharged through a tall stack. The stack height is chosen so that the maximum concentration of radioactivity at ground level is less than the permissible value.

The design of the gas-disposal system is strongly influenced by the amount of long-half-lived Kr^{85}

produced in the HRR. This isotope may be produced at a rate of about 600 curies per day. Because the half life is so long (about 10 years), it is impossible to significantly reduce the activity by any reasonable holdup period. Thus the adsorption beds are designed to remove the shorter-lived activities, and the stack is designed to discharge safely the Kr^{85} produced. A stack height of 250 ft is estimated to be adequate.

3.4 STARTUP AND SHUTDOWN

In the recommended method of startup,⁴ the high-pressure system is filled with heavy water and then preheated to the desired critical temperature by injecting steam into the main heat exchangers. The reactor is then brought to critical by pumping in concentrated fuel solution. The final operating temperature is reached under reactor power. Gas evolution will not be a problem if a sufficiently high critical temperature is selected. Figure 7 indicates the reactor power at which gas bubbles would start to appear for a given average reactor temperature and overpressure. Also shown is the power required to make up natural heat losses from the system. The reactor power at which bubbles would start to appear for other overpressures is proportional to the overpressure. It is apparent from the graph that high critical temperatures will be required if the reactor is to operate at powers in excess of the natural heat losses. The amount of excess power available determines the instantaneous rate at which the reactor can be brought from critical to operating temperature without gas evolution, although it may be possible to use gas separators in the outlet lines to allow startup at lower temperatures. This startup method is very safe nuclearwise but is time consuming because of limitations on the

⁴P. R. Kasten, *Control and Start-up of HRR*, ORNL CF-56-5-124 (May 18, 1956).

UNCLASSIFIED
ORNL-LR-DWG 18352

Fig. 7. HRR Power Required to Form Gas Bubbles (200-psi Overpressure).

maximum rate of temperature rise due to thermal stress; it might also be uneconomical to procure a large-capacity boiler for preheating the large fluid inventory.

Instead of using steam to preheat the high-pressure system, most of this energy requirement may be supplied by fission-product decay heat. It is proposed that during each routine shutdown sufficient fuel solution be retained in the high-pressure system to keep the system above a minimum specified temperature during the shutdown period. Radiolytic gases generated can be removed by a continuous steam vent. Decay heat in excess of requirements for heat losses and venting can be removed by generating steam in one or more of the main heat exchangers at a low controlled pressure. At the time of reactor startup the high-pressure system is filled with heavy water that has been preheated to the system equilibrium temperature. The reactor is made critical at this temperature by pumping in concentrated fuel solution.

This method of operation eliminates part of the thermal cycle on the high-pressure system and may be significant in preventing leaks in the system. Another advantage is the reduction in time required to go critical, since the high-pressure system already contains part of the critical mass required. This method can be made sufficiently safe nuclear-wise by restriction of the fuel and heavy-water holdup in the high-pressure system.

3.5 CRITICAL TEMPERATURE

The selection of a critical temperature for the HRR was based on a study of heat losses and venting requirements for the high-pressure system during shutdown. The holdup of fissionable material in the high-pressure system is in the return piping to the reactor; none is in the reactor proper. For the purpose of the initial feasibility study it was assumed that if the holdup of fissionable material in the return piping is less than the critical mass of the reactor itself at the holdup temperature, the system will be adequately safe. The possibility of an incident occurring during shutdown which could transfer all fissionable material to the reactor and establish a new critical geometry with a smaller fluid and fissionable-material inventory is extremely remote. Detailed nuclear calculations may later show that larger holdup inventories are permissible. In this event the idea of partial holdup of the fuel solution during periodic shutdown will appear even more attractive.

The critical-temperature selection for the HRR is based on an operating cycle of two weeks and a shutdown period of two days. The results apply directly to longer operating cycles. Figure 8 shows the amount of heat loss from the high-pressure system and the amount of fissionable material which must be held up in the system after

UNCLASSIFIED
ORNL-LR-DWG 18353

Fig. 8. Fuel Holdup in High-Pressure System of HRR.

shutdown in order to maintain a given equilibrium temperature over a two-day period.⁵ Also shown is the amount of fissionable material required to make the reactor itself go critical at a given equilibrium temperature. An equilibrium temperature of 374°F was selected for the HRR. At this temperature the power required to increase the reactor temperature is about 1 Mw. This is well below the power level at which gas bubbles start to appear. At lower critical temperatures gas evolution becomes a problem. At higher critical temperatures the large holdup of fissionable material during shutdown becomes a nuclear-safety problem.

3.6 THIMBLE-RUPTURE HAZARD

In order to maintain an equilibrium temperature of 374°F during shutdown, it is necessary to keep the high-pressure system at the saturation pressure, 182 psia. The existence of a net positive pressure on the outside of the thimbles during shutdown may constitute a hazard. The thimbles are adequately strong for this condition without internal pressurization, but perhaps some unusual circumstance might cause a rupture. In the event that a rupture should occur, it can be confined to the thimble if some type of closure is provided on the inlet port. If no closure is provided, the vapor expansion must be contained by the hot cell. This would require exclusion of personnel from this area if any thimble inlet ports are open.

3.7 PROCESS DESIGN

Most of the HRR process design is similar to that proposed for other one-region homogeneous reactors. However, no gas separator is needed. A continuous fluid letdown of about 4 to 10 gpm is required to maintain constant liquid level in the high-pressure system. This is due to the pressurizer and circulating-pump purges and the fuel feed stream. The dump-tank design is slightly different also, in that one of the dump tanks is isolated from the remainder by a valve. This allows the fluid volume that is displaced from the high-pressure system during startup to be isolated from the concentrated fuel feed solution in the dump tanks. Preventing dilution of the fuel feed solution greatly reduces the startup time required.

⁵C. Michelson, *Maintaining High Pressure System of the HRR at Temperature After Dump*, ORNL CF-56-7-38 (July 10, 1956).

3.8 NORMAL OPERATION

During normal operation of the HRR the letdown tank remains empty. Fluid letdown from the high-pressure system, blowdown from the high-pressure steam generator, and a vent stream from the pressurizer are transferred to the dump tanks. Fission-product and radiolytic gases pass upward through the entrainment separator and recombiner system during a letdown. Fuel feed for the high-pressure system is taken from the dump tanks. Condensate feed is taken from the condensate tanks.

Prior to shutdown of the HRR to change experiments, the reactor load is removed and the fuel solution is circulated for several minutes to ensure complete recombination of the radiolytic gases. The main circulating pumps are then stopped, the steam overpressure of 1400 psia is released, and the dump valve is opened. At this instant the high-pressure system is at 577 psia.

During periodic shutdown about 35,000 liters of fuel solution is dumped from the high-pressure system. Fission-product decay heat from the remaining 11,000 liters keeps the high-pressure system at about 374°F. An average continuous evaporation rate of approximately 250 lb/hr is required in the high-pressure system to dilute and remove the radiolytic gases being generated. Additional evaporation occurs to make up natural heat losses from the system. Decay heat in excess of the requirements for heat losses and venting is removed by generating steam in one or more of the main heat exchangers at a low controlled pressure.

The fuel solution transferred to the dump tanks is slowly cooled to 212°F. The vapor phase is condensed in the recombiner condenser and transferred to the condensate storage tanks. The non-condensables are vented to the cold traps and fission-product beds.

As the evaporation process continues, the fuel-solution concentration in the dump tanks increases. When the concentration reaches 100 g/liter, as determined by the fuel sampling system, a flow of condensate from the recombiner condenser is started to maintain approximately constant fluid inventory in the dump tanks. This may be accomplished by a liquid-level controller coupled to a valve in the condensate return line. A preliminary study of the chemistry and corrosion problems associated with concentrations of 100

g/liter at 100°C and fission-product power densities of about 200 w/liter has been made.⁶ There appears to be no serious chemistry problem, such as solution stability, but a corrosion problem may exist. However, the concentrated fuel solutions are retained in the dump tanks for relatively short periods of time; so the integrated corrosion may still be small. Titanium may be required for the process lines and equipment transferring the concentrated fuel solution to the high-pressure system.

During the concentration process in the dump tanks, vapor is let down from the high-pressure system to carry off radiolytic gases. This loss of fluid from the high-pressure system is made up by an occasional transfer of fluid from the condensate storage tanks. The condensate tanks are equipped with evaporators for this purpose. It is not mandatory that any definite amount of fluid be maintained in the high-pressure system.

At the time of reactor startup the high-pressure system contains a holdup inventory of fissionable material and heavy water. The dump tank contains the concentrated fuel solution, and the condensate tanks contain the heavy-water inventory for filling the high-pressure system. The letdown tank is empty.

Prior to startup of the HRR the condensate tanks are valved off and then heated by the evaporators to sufficient steam overpressure to force the condensate into the high-pressure system. Before the main circulating pumps are started, 200-psi steam overpressure is applied to the system in order to prevent boiling due to nuclear surges and to prevent radiolytic-gas evolution. After circulation begins, the fuel feed pumps are started. While the reactor is being brought to critical, the letdown stream from the high-pressure system is held up in the letdown tanks. This prevents dilution of the concentrated fuel feed solution in the dump tanks. About 70 min is required to make the reactor critical at 374°F.

After the reactor goes critical, it is brought to operating temperature under its own power. The steam overpressure is slowly increased to 1400 psia. The volume expansion that occurs during heating of the fuel solution will necessitate ejecting fluid into the letdown tanks. About 47 min is required to bring the reactor from critical to full-power operation.

⁶R. A. McNees, personal communication.

Table 7 shows the assumed condition of the reactor system at the time of reactor startup, at the time the reactor reaches critical, and at full-power operation. Table 8 shows the important HRR startup parameters.

When the normal operating condition has been reached, the letdown tank is emptied and the condensate tanks are filled with the excess condensate. The dump-tank concentration is reduced, since only a small continuous fuel feed is required to make up for fuel burnup and letdown.

No xenon override problems exist with this operating procedure, since most of this gaseous poison is removed in the off-gas system whenever the steam overpressure is released.

TABLE 7. HRR INVENTORIES

	Startup (kg)	Critical (kg)	Full Power (kg)
High-pressure system			
D ₂ O	9,800	44,800	40,700
U ²³⁵	8.8	28.0	36.8
Letdown tank			
D ₂ O	0	3,970	10,900
U ²³⁵	0	1.8	7.4
Dump tanks			
D ₂ O	4,900	2,120	210
U ²³⁵	37	16	1.6
Condensate tanks			
D ₂ O	38,300	2,110	1,190

TABLE 8. HRR PARAMETERS

	Startup	Critical	Full Power
Average temperature, °F	374	374	482
Pressure, psia			
Saturation	182	182	577
Total	182	390	1400
Time from startup, min	0	70	117
Rate of temperature rise, °F/min	0	0	2.3

4. EXPERIMENTAL FACILITIES

4.1 TYPES OF EXPERIMENTS

The in-pile experimental facilities are to be designed for maximum versatility so that a wide variety of in-pile experiments can be accommodated. The types of experiments which may be visualized can be classified as follows:

(a) **Engineering.** — Engineering experiments are typified by the various in-pile homogeneous-circulating-fuel loop facilities which are designed and operated by ORNL for the purpose of determining the effect of irradiation on corrosion and fuel-solution stability. In general, they consist of a scaled-down version of an essentially complete reactor system which is compact enough to be inserted into existing experimental facilities. Successful loop experiments have been designed to function within 6-in.-dia holes in the LITR and the MTR.

(b) **Irradiation of Materials.** — In order to successfully design power reactors, it is mandatory that information concerning the effects of high fluxes of fast and thermal irradiation on properties of materials be secured as quickly as possible. In addition to providing facilities for irradiating materials, the experimental facilities of the HRR also could be used for isotope production.

(c) **Pure Research.** — The facilities of the HRR may be designed to provide collimated beams of high-intensity neutron and gamma rays for conducting research into the fundamental properties of matter.

4.2 DESCRIPTION OF FACILITIES

Six horizontal in-pile thimbles equispaced on the midplane of the reactor and one vertical nozzle opening from the top of the reactor are included in the design. Figure 9 shows the location of the thimbles relative to the containment vessel and shielding. For convenience in performing experiments, the reactor is located such that the center lines of the experimental thimbles are from 30 to 36 in. above the operating-floor level.

The six penetrations for the experimental thimbles through the shell of the reactor vessel are conservative in number as far as the strength of the shell is concerned. It is possible to increase the number of thimbles and perhaps rearrange them on an unsymmetrical spacing. As shown by Fig. 9, piping to the heat exchangers must be run

from one side of the reactor, presumably through or adjacent to one of the hot-cell working areas; hence, this area may not be usable for experiments. This area may be used, however, to contain the pressurizer or other items which must be adjacent to the reactor but removed far enough from the reactor cell to permit maintenance.

4.3 THIMBLE DESIGN

The following were design considerations in an investigation⁷ to determine the suitability of a solid-wall cylindrical shell type of thimble construction subjected to the proposed reactor operating conditions:

1. Thimble material is Zircaloy-2, necessary because of the requirement of low neutron-absorption cross section.

2. The thimbles are required to withstand intense internal heating resulting from the absorption of gamma rays. This energy release is estimated to be as high as 4 w per gram of Zircaloy-2.

3. The thimble shell is required to support externally a reactor operating pressure of 1400 psia (no reliance is placed on internal coolant pressure for shell strength). A requirement of the design is that the shell shall have adequate strength against both plastic yielding and buckling types of failure.

4. Space requirements for experiments indicate a thimble size as large as possible within design feasibility.

5. Boiling heat transfer from the thimble to the fuel solution shall be avoided because of acceleration of corrosion.

Conclusions reached were:

1. A 6-in. internal diameter is near the maximum allowable thimble size without a requirement of internal pressurization of the shell. The size is limited by design criteria for maximum allowable thermal stress and adequate buckling strength.

2. The minimum thickness for a 6-in.-ID thimble is $\frac{3}{4}$ in., limited by buckling.

3. The maximum thickness for a 6-in.-ID thimble is 1 in., limited by thermal stress.

4. Up to $\frac{1}{4}$ in. may be added to the minimum thickness of a 6-in.-ID shell as a corrosion allowance.

⁷R. F. Hughes, *The HRR Thimble Design Problem*, ORNL CF-56-6-175 (June 27, 1956).

Fig. 9: HRR Pressure-Vessel Shielding and Experimental Facilities; Conceptual Design.

5. The ratio of internal diameter to thickness for thimbles less than 6 in. in inside diameter should be 8 to 1 plus a thickness allowance for corrosion.

6. Internal coolant temperatures for a 6-in.-ID $\frac{3}{4}$ -in.-thick shell should be held below 200°F to help attain convection (not boiling) heat transfer to fuel solution.

Thimbles which house flux converters will be subjected to an intensity of heating much greater than the 4 w per gram of Zircaloy-2, on which the foregoing analysis was based. Design measures that have been suggested to limit thermal stress in these thimbles are:

1. Reduce the diameter of the shell to below 6 in.

2. Reduce the wall thickness and rely on high internal coolant pressure for shell strength.

3. Fabricate the shell in laminated layers, one shrunk over the other.

4. Provide internal cooling in the thimble wall.

The latter three design suggestions also offer a possible means of securing thimbles larger than 6 in. in inside diameter.

4.4 DESIGN PROPOSALS

Figures 10 and 11 show two possible designs for the experimental thimbles. From the standpoint of simplicity of experiment design, the construction shown by Fig. 10 is recommended. With this design, however, the wall of the thimble must with-

stand the full design pressure of 1400 psi, and a suitable joint between the Zircaloy-2 thimble and the steel pressure vessel must be developed to withstand this pressure also.

The design shown by Fig. 11 is proposed to alleviate the problems of designing the thimble and joint for the full pressure differential. With this design, the external pressure tending to collapse the thimble can be balanced by internal pressure, with a resultant thinner wall, and the type of balanced-pressure joint developed for use in the HRT Zircaloy-to-stainless-steel connection may possibly be used. However, a balanced-pressure thimble design of this type will complicate the design of the experiments. In the event

UNCLASSIFIED
ORNL-LR-DWG 18355

Fig. 10. Thimble Design (1400-psia External Pressure).

UNCLASSIFIED
ORNL-LR-DWG 18356

Fig. 11. Alternate Thimble Design - Balanced Pressure.

of failure of the thimble, the balancing chamber may be expected to retain safely the contents of the reactor, which otherwise would discharge to the cell.

The problem of designing a zero-leakage mechanical joint between materials with widely different coefficients of thermal expansion such as exist between Zircaloy-2 and stainless steel has been analyzed in connection with the core-tank design of the HRT.⁸ However, the joint specified and developed was for a comparatively small

⁸C. L. Segaser, *Zircaloy-to-Stainless Steel Transition Pipe Joint Design Study*, ORNL CF-55-12-72 (Dec. 5, 1955).

pressure difference. The design of a joint of this type for a 1400-psi pressure difference will present problems of a different order of magnitude and will require considerable development to be successful.

Experiments which may be carried out in the thimbles will result in the handling of large volumes of radioactive material. In recognition of this fact, the containment vessel surrounding the reactor should be large enough to allow adequate working area adjacent to the thimbles to contain shielded coffins and other necessary remote-handling equipment. The hot-cell areas shown in Fig. 9 would be designed for these experiments.

5. MECHANICAL DESIGN OF REACTOR COMPONENTS

5.1 HIGH-PRESSURE SYSTEM

(a) **Reactor Vessel.** — The reactor vessel for the HRR is essentially a 10-ft-ID spherical pressure vessel designed according to the rules of the ASME Unfired Pressure Vessel Code, with consideration given to the special problems introduced by the heating of the shell from radiation absorption and by the necessity of penetrating the shell for insertion of experimental thimbles. The vessel shown in Fig. 12 may be fabricated of a carbon-steel base material with a stainless steel cladding on all surfaces exposed to fuel solution. The design criteria are that the vessel must contain the fuel solution at 1400 psia internal pressure and at a design temperature of 600°F.

The fuel solution enters the vessel through two 24-in. nozzles welded to the bottom half of the vessel, flows upward through the vessel, and exits through two 18-in. nozzle connectors in the top. A diffuser screen, which in this design consists of a part of the thermal shield, is placed at the entrance to the reactor vessel to distribute flow throughout the sphere.

Nozzle connections at the entrance to the reactor vessel are sized for an allowable fluid velocity of 10 to 15 fps. At the exit, the nozzles are sized for a fluid velocity of 30 to 40 fps. Fluid velocities in the reactor vessel should be under 3 fps.

Figure 13 shows the calculated total tangential stress for the pressure vessel at steady-state operation (500 Mw and 1400 psi internal pressure) as a function of wall thickness and thermal shielding. The maximum stress, at the inside shell surface, results from internal pressure and irradiation heating.

For type A-212-B steel having an allowable working stress of 17,500 psi, a minimum wall thickness of about 3.30 in. is required with 5 in. of internal thermal shielding. The thermal shielding may be of laminated construction as indicated on Fig. 12. The inside of the carbon-steel pressure shell and all exposed surfaces of the thermal shield are clad with stainless steel to inhibit corrosion.

A stainless steel blast shield is placed around the reactor vessel to contain fragments of the vessel in the event of a brittle failure, and cooling

coils are wrapped around the blast shield to control the pressure-vessel temperature.

There is little doubt as to the feasibility of purchasing a pressure vessel such as the one required for the HRR, since pressure vessels of larger diameter and for higher pressures have been designed and fabricated.

(b) **Heat Exchangers.** — Preliminary specifications for heat removal from the reactor system are listed in Table 9. Although these specifications are not the optimum design conditions, they are believed to be very near the optimum, from comparison with

TABLE 9. HRR STEAM-GENERATOR SPECIFICATIONS

Reactor fluids, forced circulation	
(tube side)	
Inlet temperature, °F	527
Outlet temperature, °F	437
Flow rate, lb/hr	2,730,000
Pressure, psia	1400
Velocity through tubing, fps	10
Steam, natural recirculation (shell side)	
Generation temperature, °F	417
Pressure, psia	300
Generation rate, lb/hr	351,600
Heat load, Btu/hr	284,300,000
Heat load, Mw	83.3
Steam generator	
Number of $\frac{3}{8}$ -in. 18 BWG tubes	3280
Pitch, in.	$\frac{5}{8}$, triangular
Effective length of tubing, ft	25.9
Heat-transfer surface, ft ²	8330
Shell internal diameter, in.	38 $\frac{1}{2}$
Shell thickness, in.	1 $\frac{7}{8}$
Tube-sheet thickness, in.	5
Reactor fluid connections	14-in. sched- 100 pipe
Number of risers	12
Number of downcomers	6
Size of risers and downcomers	5-in. sched- 40 pipe
Steam drum	
Internal diameter, in.	36
Length, ft.	16
Wall thickness, in.	1 $\frac{3}{4}$
Height above generator, ft	15

Fig. 12. HRR Pressure Vessel – Conceptual Design.

Fig. 13. Total Stress vs Wall Thickness, HRR Pressure Vessel.

previous studies made on similar designs. Six heat exchangers of 83.3-Mw capacity each are required to dissipate the 500 Mw of heat generated in the reactor.

The steam generator in the present design consists of a lower vaporizing shell connected to a steam drum at a suitable elevation to promote natural circulation by means of risers and downcomers welded to the shells. Other possible types of steam generators may be considered, however, such as the bent-tube single-shell type and the once-through type, but detailed analysis of these types will be deferred until an optimization study can be made. An evaluation⁹ of the feasibility of large-capacity steam-generating heat exchangers indicates that units of this type can be designed and procured.

(c) Circulating Pumps. — Large circulating pumps for homogeneous-reactor service will fall into two

capacity ranges: 4000- to 5000-gpm pumps designed for STR, SIR, and SFR use and 16,000- to 17,000-gpm pumps designed for PWR use.

For the 4000-gpm pumps designed to circulate 2000-psi water, development has advanced to the point where operational feasibility has been conclusively demonstrated and operational reliability has been reasonably well established. The development of 16,000-gpm pumps is not so advanced. The high-pressure pumps, in general, lend themselves to the modifications and material changes necessary for suitable homogeneous-reactor circulating pumps. Material changes include using higher quality stainless steels and higher quality welding, replacing certain parts with titanium, and probably replacing the Inconel stator and rotor cans with stainless steel.

(d) Pressurizer. — Because the technology of vapor pressurization is more advanced than that of gas pressurization, this mode was selected for this HRR study. Gas pressurization, however, has many advantages, and should be given further consideration before the final design is made.

The pressurizer surge chamber, constructed of 24-in. sched-100 pipe and arranged to fit into the available space above the reactor, provides the necessary 1500 liters of surge volume.

Steam for the HRR pressurizer is developed in a small high-pressure steam generator physically separated from the pressurizer surge chamber. Space limitations and accessibility problems make this separation desirable. This generator is a shell-and-tube heat exchanger with a shell of an 8-in. sched-80 pipe about 2 to 2½ ft long. Fifty ½-in. 18-gage tubes, 1 ft 9 in. long, provide the necessary heat-transfer area. This generator will produce 60 lb/hr of steam at 1400 psia (587°F) on the shell side. An auxiliary boiler to provide 1700-psia (613°F) steam on the tube side is required. Heating by condensing steam offers the advantage of uniform wall temperatures, which are limited to a definite maximum temperature. The temperature of the outside surface of the tubes would be about 589°F for this situation. With this low wall temperature and a continuous blowdown equal to 50% of the generation rate, a serious problem due to the impurities in the feed water would be less likely. Careful design and operation will be required to keep the solids buildup low in this generator.

⁹Foster Wheeler Corporation, Job No. 8-51-8526, final report Dec. 15, 1956; Job No. 8-51-8529, final report Nov. 15, 1956.

An economizer is placed in the blowdown line upstream of the control valve. This economizer is a double-pipe heat exchanger 8 ft long. The blowdown stream passes through the inner pipe, which is a $\frac{3}{4}$ -in. sched-40 pipe, and the feed stream passes through the annular space between this pipe and the outer pipe, which is a $1\frac{1}{4}$ -in. sched-80 pipe.

This economizer is used to recover some of the heat from the liquid in the blowdown stream and to cool this liquid before it passes through the blowdown valve. A combination of a high temperature and impurities could cause an excessive corrosion rate in this valve.

The feed-water pump for the steam generator is a Pulsafeeder-type pump with a variable-speed drive. The feed rate is controlled by the liquid level in the steam generator, as determined by a liquid-level controller. A block valve in the blowdown line allows a tight shutoff in case of a pump failure.

The steam generator must produce a sufficient amount of steam to make up heat losses of the pressurizer surge chamber, reactor level controller, and connecting piping. The surge chamber has about 138 ft² of heat-transfer surface at a temperature of 587°F, and the ambient-air temperature is assumed to be 150°F. The radiation heat-transfer coefficient from the surface is 2.1 Btu/hr·ft²·°F, and the natural-convection coefficient is 1.33 Btu/hr·ft²·°F. Assuming 90% insulation efficiency, the resulting heat loss is 6.1 kw. Insulated heat losses in the liquid-level controller and the connecting piping and miscellaneous losses increase the total heat loss to 10 kw. The steam generator must supply a minimum of 60 lb/hr of saturated steam at 1400 psia to make up this heat loss. In the design described here, it is assumed that no venting is required for the normal operation of the HRR. If venting should be required, additional steam would have to be produced by this generator. The reactor develops 620 lb-moles of D₂ per hour at full power. The maximum safe concentration of D₂ in the pressurizer is 5 mole %. If it is assumed that 0.1% of the D₂ produced in the reactor enters the pressurizer before being recombined, a continuous vent of 125 lb/hr of steam is necessary for safety. The steam generator would then have to be designed to supply 185 lb/hr of steam.

5.2 LOW-PRESSURE SYSTEM

The HRR dump tanks are designed to store safely the entire fluid inventory of the high-pressure system in case of an emergency dump. For normal plant operation the tanks would serve as a storage area for part of the fuel inventory during periodic shutdown and as a reservoir for concentrated fuel solution during startup and full-power operation. A total dump-tank capacity of 65,000 liters is provided. This gives a minimum excess volume of 25%.

A dump-tank geometry has been selected which can be considered safe for the HRR operating conditions. These tanks are not so limited in their dimensions that a chain reaction cannot be started regardless of the fissionable-material concentration. Such an arrangement would be expensive and cumbersome. The alternative is to limit the linear concentration of fissionable material in the tanks by providing adequate mixing. Criticality data have been published¹⁰ for very long cylinders coated with a layer of cadmium and having an infinite H₂O reflector. An extrapolation of these data indicates that a linear concentration of less than 0.4 kg of fissionable material per foot should be safe in a 36-in. cylinder. Since the maximum allowable concentration of fissionable material per foot of pipe is independent of the quantity of solvent, any desired degree of concentration by means of evaporation is permissible.

A total dump-tank capacity of 65,000 liters requires 325 ft of 36-in.-ID (200 liters/ft) tanks. The uniform fissionable-material concentration after a complete dump would be 0.136 kg/ft (44.2 kg); 15,000 liters of this volume would be the letdown-tank capacity. If all the fissionable material were in the dump tanks, excluding the letdown tanks, the uniform material concentration would be 0.177 kg/ft.

It should be noted that larger dump-tank diameters might be used if detailed calculations verify their safety. For this initial study it was felt that a large factor of safety should be incorporated into the design, particularly in view of the limited extent of the data and uncertainties as to the degree of fuel-solution uniformity which might be attained in such large fluid inventories during the evaporation process.

¹⁰S. Visner and L. C. Noderer, *Criticality of U²³⁵. H₂O-D₂O Systems in Cylindrical Geometry*, ORNL CF-54-5-170.

(a) **Design Temperature and Pressure.** — If the large initial decay heat were to be quickly removed from the system, the final equilibrium pressure in the dump tanks after an emergency dump of the high-pressure system would be lower than the saturation pressure of the high-pressure system at the average operating temperature (482°F, 577 psia). If this decay heat were not removed, perhaps because of loss of cooling water to the recombiner-condenser or of feed water to the heat exchangers, a net pressure rise during a dump could result. For this initial HRR design study it was assumed that no transfer of heat to or from the combined high- and low-pressure systems and that no generation of heat within the combined system due to volume heat sources would occur during a dump. Thus the final equilibrium pressure in the dump tanks after an emergency dump of the high-pressure system is assumed to be that due to an adiabatic expansion of the fuel solution in the high-pressure system into the combined volume of the high-pressure system and dump tanks. No allowance is made for the volumes in the entrainment-separator, recombiner, recombiner-condenser, or condensate-storage-tank. The characteristics of light water are used for this calculation. The final equilibrium pressure is calculated to be 475 psia; on this basis, a design pressure of 500 psia is recommended at a design temperature of 600°F.

(b) **Dump-Tank Construction.** — Type 347 stainless steel plate is recommended for fabricating the HRR dump tanks. An inside diameter of 36 in. has been selected for the tanks. The wall thickness required according to the Code is 0.65 in.; a corrosion allowance of 0.1 in. is specified, resulting in a wall thickness of 0.75 in.

(c) **Dump-Tank Configuration.** — A tentative location has been selected for the dump tanks in the reactor plant (Fig. 15). Until the space limitations become known, however, the best type of configuration cannot be determined.

Evaporators are required on the dump tanks to take care of startup requirements when fission-product decay heat is not available and perhaps also to agitate the fuel solution for better mixing. The type and location of evaporators will be determined after the dump-tank space limitations are known.

The letdown tanks, separated from the rest of the dump tanks by a valve, are 75 ft long and have a volume of 15,000 liters. The 15,000-liter volume

is needed to contain the material let down from the high-pressure loop during the startup operation in order to prevent it from mixing with the concentrated solution in the remaining dump tanks.

(d) **Condensate Storage Tanks.** — The HRR condensate storage tanks are designed to hold a fluid volume equal to that of the high-pressure system less the fuel-volume holdup in the high-pressure system. A minimum excess volume of 11% is provided, giving a total condensate storage capacity of 40,000 liters. Since it would be possible for fissionable material to fill the condensate tanks through a leak in the transfer valve to the high-pressure system, it is necessary that the tanks be restricted to a safe diameter. The worst condition to assume is one in which the entire fissionable-material inventory of the high-pressure system becomes uniformly distributed in the condensate tanks. On the basis of criticality studies made for the dump tanks, it appears necessary to restrict the condensate tanks to 36 in. in inside diameter. Larger tanks might be used if more detailed calculations verify their safety.

A condensate storage capacity of 40,000 liters requires 200 ft of 36-in.-ID tanks. A uniform fissionable-material concentration of up to 0.222 kg/ft could appear in these tanks if all the fissionable material collected there.

Another 2500-liter condensate tank is required to collect the D₂O from the cold traps and to store separately the purge water for the startup operation during the preheating of the condensate for the loop. A valve again is used to separate this tank from the rest of the condensate tanks.

(e) **Condensate-Tank Construction.** — Since it would be possible for a high-pressure leak to develop in the condensate transfer valve, the condensate tanks are designed for the same pressure and temperature as the dump tanks, 500 psia and 600°F. The same wall thickness (0.75 in.) is required because the same inside diameter is specified. The corrosion allowance for the condensate tanks can be smaller, but this distinction has not been made. No particular configuration is required because there is no mixing problem.

(f) **Feed Pumps.** — Several positions are shown on the flowsheet, Fig. 5, where fuel solution or D₂O must be pumped from containers at atmospheric pressure to lines at the system pressure of 1400 psia. These applications include the supply of purge water (D₂O) to the circulating pumps,

feed water (D_2O) to the pressurizer steam generator, and concentrated fuel solution to the reactor system. A zero-leakage pump is required for these applications.

A Scott & Williams P-14 Hydropulse pump, with check valves removed, supplying pulsations through an intermediate water line to two or more Lapp-type Pulsafeeder heads meets the requirement of a zero-leakage pump unit. A system of two Lapp Pulsafeeder heads driven by one Hydropulse pump unit has a maximum output of about 4 gpm when pumping to 2000 psi. However, unit combinations of this type tested in high-pressure simulated reactor loops have exhibited very erratic behavior and a high probability for failure. Failures have occurred in the pulsator housings, in the Pulsafeeder and purge-pump suction check valves, and in the diaphragms. Although certain precautions and preventive measures may be taken to improve operation of the pulse-type units, the situation in regard to suitable high-pressure feed pumps for the HRR is grave enough to warrant an intensified investigation of this problem.

(g) Entrainment Separator. — An entrainment separator is required to separate the entrained liquid and fuel from the steam coming from the dump tank. This separation must be complete in order to eliminate the loss of fuel and to prevent any wet material from entering the recombiner.

This separator, which is similar to those designed for the HRT and TBR, contains a 12-in. baffle section followed by a 12-in. coarse-wire-mesh section and a 12-in. fine-wire-mesh section. These beds are housed in a 4-ft 8-in. cylinder with an 18-in. sched-80 inlet pipe and a 14-in. sched-80 outlet pipe. Vanes are placed in the inlet pipe in order to give the fluid a swirling motion, which will help to separate some of the entrained fluid.

This design permits a superficial velocity of 13.8 fps at a 9-Mw heating rate and 1.53 fps at the 1-Mw heating rate. The efficiency of the separator is good in this range, and the separation should be quite complete.^{11,12} The highest separation efficiencies are in the velocity range of 4 to 13 fps.

Provision should be made to backwash the separator with D_2O in order that the collected

solids may be washed out of the wire-mesh beds. This could be done with cooling coils or by a D_2O inlet at the exit of the separator.

(h) Recombiner. — The recombiner must contain sufficient catalyst to assure a complete recombination of the deuterium and oxygen gases. A space velocity of $420,000 \text{ hr}^{-1}$ is assumed to be sufficient for 100% recombination¹³ and a safety factor of 2.5 is used for the volume of catalyst.

An emergency dump of the reactor fluid would not be instantaneous but would take place over a finite time. At the end of 10 min after shutdown, heat would be produced at the rate of 9 Mw. At these conditions, a minimum catalyst bed volume of 5 ft^3 is required. This bed contains $\frac{1}{8} \times \frac{1}{8}$ in. cylindrical pellets of 0.37% platinized alumina.

This packed bed is a cylindrical annulus with a 24-in. ID, 24-in. length, and 4-in. thickness. The inner and outer cylindrical walls are screened to allow the steam and gases to enter the bed and the superheated steam to leave the bed. The inlet and outlet pipes are 14-in. sched-80, which limits the steam and gas velocities to a maximum of 250 fps. Care must be taken that the recombiner bed be kept dry, as it is ineffective when wet. The heat of reaction is usually sufficient to ensure dryness, but an auxiliary steam heater should be provided on the inlet pipe.

(i) Recombiner-Condenser. — The hot D_2O vapor leaving the recombiner is condensed in a water-cooled condenser. (The heat load required for this condenser is very close to the loads used for the TBR.) This condenser is a standard two-pass heat exchanger with a floating head. The D_2O steam is on the shell side, and the cooling water is in the tubes. Table 10 gives the approximate specifications for the condenser.

5.3 CONCEPTUAL LAYOUTS OF REACTOR COMPLEX

Preliminary conceptual layouts of a reactor complex for the proposed homogeneous research reactor plant are shown by Figs. 14, 15, and 16. These layouts show only the relation of the items pertaining to the nuclear-reactor components. The relations of the turbogenerator facilities and other facilities to the reactor plant are shown on a general site plan, Fig. 17.

¹¹C. L. Segaser, *HRT Entrainment Separator Design Study*, ORNL CF-54-7-122 (July 23, 1954).

¹²O. H. York, *Chem. Eng. Progr.* 50(8), 421 (1954).

¹³R. E. Aven, ORNL CF-54-11-1 (Nov. 1, 1954) (Classified).

TABLE 10. RECOMBINER-CONDENSER SPECIFICATIONS

Heat load	5.75×10^5 Btu/min
Condensing temperature	212°F
Log mean temperature difference	89°F
Heat-transfer coefficient	230 Btu/hr·ft ² ·°F
Coolant flow rate	9670 lb/min
Coolant inlet temperature	90°F
Coolant outlet temperature	1500°F
Tube size	$\frac{1}{2}$ -in. BWG 20
Tube pitch	$\frac{5}{8}$ in., triangular
Tube length (between tube sheets)	18.6 ft
Number of tubes per pass	350
Number of passes	2
Coolant velocity in tubes	7.4 fps
Shell inside diameter	18 $\frac{1}{2}$ in.
Material of construction	Type 347 stainless steel

Figure 14 is a plan view of the reactor complex; it indicates the general relation of the reactor pressure vessel and its auxiliaries to the heat exchangers and circulating pumps. The requirement that the reactor provide experimental facilities necessitates that the reactor pressure shell containing the experimental thimbles be placed in a separate shielded containment vessel. In order to provide adequate working area adjacent to the reactor experimental thimbles, it is estimated that the containment vessel should be approximately 60 ft in diameter. This size of vessel allows the working areas to be shielded for limited access to the experiments when the reactor is shut down. Approximately 6 ft of concrete is shown around the reactor area, with an additional 3 ft, pending precise shielding calculations, shown around the periphery of the containment vessel. Thus the total biological shielding may be 9 ft or more in thickness of high-density concrete. One or more of the areas, however, may have all the biological shielding arranged adjacent to the reactor cell to facilitate experimentation where total access to the experimental thimble is required.

Sectional elevations of the reactor complex are shown by Figs. 15 and 16. A condition of the

reactor layout is that the arrangement of the heat exchangers relative to the reactor vessel is such that natural circulation through the system will be promoted in the event of pump failure. Since the center line of the reactor vessel should be located at 30 to 36 in. above the operating-floor level for convenience in experimentation, the containment vessels for the heat exchangers and circulating pumps are above ground. The containment vessel for the reactor is visualized as a vertical cylindrical tank. Two horizontally mounted containment vessels, each 60 ft in diameter, are required to house the heat-exchanger equipment. Adequate biological shielding must be placed around these vessels. The dump tanks are shown on these layouts to be directly under the heat-exchanger containment vessels. A scheme for limited access to those portions of the dump-tank system which will require periodic maintenance, such as dump valves, must be developed.

Maintenance of equipment in the highly radioactive system of a circulating-fuel homogeneous reactor complex has been and continues to be the most serious problem in the layout of a practical system. Either the equipment must be designed and arranged so that it can be maintained in place or so that it can be removed to an area for maintenance and be replaced with a new unit while this maintenance is being done. Removal and replacement has been the general scheme considered to date. Both dry and underwater removal methods have been investigated, but both schemes present difficult design problems. For large systems such as the HRR, however, it is believed that dry maintenance with appropriate shielding and built-in special tooling may have the best chance of ultimate success. This maintenance philosophy has been followed in the layouts presented herein.

Maintenance of equipment in the reactor compartment is expected to be largely confined to the reactor auxiliary equipment, such as the pressurizer, and to the experimental thimbles and equipment. The reactor vessel should be designed for the life of the system, since there are no moving parts or complicated internal joints. The pressure-vessel wall and corrosion liner should be thick enough to last for the designated period.

The handling equipment in the reactor containment vessel would essentially consist of a revolving-type crane with a shielded cab for the operator and/or provision for remote operation from

30

UNCLASSIFIED
ORNL-LR-DWG 18358

Fig. 14. Proposed HRR Layout - Plan View.

31

UNCLASSIFIED
ORNL - LR - DWG 18359

Fig. 15. Proposed HRR Layout - Sectional Elevation.

32

UNCLASSIFIED
ORNL-LR-DWG 18360

Fig. 16. Proposed HRR Layout - Section A-A.

Fig. 17. Proposed Site Plan - HRR.

outside the shielded area. Commercial, remotely operated television could be utilized to expedite maintenance. Access from above to any part of the area would thus be possible. In this layout, all flanges and pipe disconnects are faced upward.

A horizontal traveling crane in each of the containment vessels for the heat-exchanger equipment is provided, also with a shielded control cab and/or

remote-operation control. Flanges connecting the circulating pumps and heat exchangers with the main piping are faced horizontally in these installations. All flanges are grouped at one end of the area, and access to the bolting is from a shielded cell by means of remotely manipulated tools. The large heat exchangers are mounted on wheeled dollies guided by tracks. After the pipe flanges

have been disconnected, the heat exchanger can be moved horizontally along the track and onto another track section which can move transversely. From this section, the heat exchanger is moved through a large air-lock type of door at the end of the containment vessel to the maintenance area. The circulating pumps are designed so that the pump impeller and motor windings may be removed vertically without removing the pump casing, which may become a part of the all-welded piping system.

During any part of the maintenance procedure, the system must be shut down and drained and the piping and equipment decontaminated as thoroughly as possible. Shut-off valves of a size and type suitable for the piping of the HRR have not been developed.

A logical requirement in the design of a circulating-fuel homogeneous reactor system is that the equipment and piping be arranged so that the volume of the external circuit will be a minimum. In the layout of the HRR, this requirement was waived owing to the need for access to the reactor for experiments. However, such a waiver is an advantage in the layout in that the longer pipe runs, with more loops and bends, result in less severe stresses due to thermal expansion. Also, a certain flexibility of equipment arrangement is afforded, which may be advantageous in equipment location for maintenance.

In the event of sudden failure of an item in the high-pressure fuel-solution system, large amounts

of highly radioactive solutions would be released. The vapors from these solutions, obviously, cannot be allowed to escape. Since the initial pressure of the system is 1400 psia and the volume is large, complete release of the fluid contents with the resultant adiabatic expansion to the containment vessel could result in relatively high pressures in the cell. The containment cell must be large in order to contain the equipment and strong enough to withstand the resultant pressure. The logical geometry of the cell is a cylinder or sphere, which allows reasonably thin cell-vessel walls and economy of construction.

To limit the radiation levels at the containment-vessel walls, the vessel is surrounded by 3 ft of high-density concrete, and an additional 6 ft surrounds the reactor. The external system requires less shielding; from 5 to 7 ft is deemed adequate from preliminary estimates.

A practical method of remotely disconnecting equipment items in the piping system is an important part of the maintenance procedures. American Standard ring-joint-flange types are commercially available and have been used in homogeneous reactor applications up to a nominal 24-in. pipe size at a 1500-psi rating. This type of pipe coupling is feasible for remote disconnects, but further development is required for the pipe sizes contemplated for the HRR. Alternate methods of remotely joining pipes, such as the use of remotely operated cutting and welding machines, should also be investigated.

6. PROCESSING OF FUEL SOLUTION

6.1 FLOW-DIAGRAM DESCRIPTION

A flow diagram depicting the method proposed for processing the fuel solution of the HRR is shown in Fig. 18. The following paragraphs explain the process principles upon which this diagram is based.

About 20 lb of fission- and corrosion-product solids will be produced each day in the reactor, based on 500 Mw of thermal energy and corrosion rates of 1 mpy (circulating system) or 10 mpy (reactor vessel) for stainless steel and 70 to 80 mpy for Zircaloy-2. These solids will be concentrated into 750 liters of fuel solution per day by the use of hydroclones with self-contained underflow pots. Two such sets of clones will be provided so that one set can be operating at reactor temperature while the other, with its pot and collected solids, is being cooled, discharged, and reheated.

In order to simplify the discussion of chemical processing, the underflow pot while operating at 275°C shall be considered as containing, in addition to D_2O , five distinct chemical species. These are as follows:

1. soluble uranium,
2. soluble fission- and corrosion-product sulfates,
3. insoluble rare-earth sulfates,
4. insoluble fission- and corrosion-product oxides and sulfates,
5. insoluble uranium.

After being cooled to 100°C the rare-earth sulfates dissolve, and a separation of liquid from solid phase, accompanied by suitable washing of the solids with D_2O , separates the major portion of the uranium from the bulk of the fission- and corrosion-product solids. These solids are slurried in D_2O to a receiver, from which D_2O is recovered. This receiver will be disconnected from the reactor system before H_2O is added and will never be used

in conjunction with a D_2O system again. The solids are then subjected to suitable storage treatment. Laboratory experiments indicate that 20 lb of such solids will contain about 0.2% uranium or about 18 g of total uranium per day. The U^{235} content will be less than 2 g, and recovery from such high-level waste seems at this time to be of doubtful value.

The solution of soluble uranium and fission and corrosion products is cooled to 30°C and D_2O_2 added. The precipitated uranium peroxide is separated from the soluble fission and corrosion products and washed with D_2O . The waste products go to the above-mentioned receiver for D_2O recovery and are then stored. (Recovery of high-activity cesium prior to combining the soluble waste with the insoluble oxides may be an attractive variation of the basic scheme.)

The precipitated UO_4 is slurried in D_2O along with D_2SO_4 and heated to about 150°C to destroy the peroxide and give UO_2SO_4 for return to the reactor.

The usefulness of the proposed scheme will be dependent upon the outcome of development and research work but is believed to be entirely feasible and represents an important and necessary step forward in the processing of homogeneous reactor fuels.

6.2 PRELIMINARY EQUIPMENT ARRANGEMENT

A preliminary equipment arrangement for the HRR chemical processing plant is presented in Fig. 19. This equipment is contained in a horizontally mounted containment vessel, approximately 24 ft in diameter by 45 ft long. The location of the chemical plant will be as close as possible to a position midway between the reactor cell and the heat-exchanger cell.

Fig. 18. Preliminary Flow Diagram of HRR Chemical Processing Plant.

Fig. 19. Preliminary Equipment Arrangement - HRR Chemical Processing Plant.

7. ECONOMIC EVALUATION

7.1 TURBOGENERATOR STATION

A preliminary investigation into the economics of generating power from a homogeneous research reactor indicates that the installation of a turbogenerator station would be preferable to the complete rejection of the energy available, under the following conditions (see Fig. 20):

Power Revenue (mills/kwhr)	Station Load Factor, Minimum
1.7	100
1.95	90
2.2	80
2.5	70
2.9	60

It should be noted that a market for "dumped power" (i.e., nonfirm power sold on an intermittent basis) exists in the Oak Ridge Area. It is estimated that a year-round average price available for such power would be in the order of 2 to 2.5 mills/kwhr.

Table 11 shows an estimate of the variation of installed and yearly charges for a turbogenerator facility as a function of the steam temperature at the throttle, as calculated from the estimated gross and net electrical power generation. Tables 12 and 13 show the heat-rejection-system costs and yearly power revenue, respectively.

7.2 COST ESTIMATE

Table 14 is a preliminary estimate of the equipment cost of the high-pressure system of the HRR including development. Using the method described by Lang,¹⁴ the total capital investment for the plant was estimated¹⁵ from the basic equipment

cost. On this basis, the total reactor plant cost (Table 15) is estimated to be \$88,380,000. To this is added the estimated cost of the turbogenerator plant (\$13,395,000),¹⁶ making the total station investment \$101,775,000 or \$830 per kilowatt of net station electrical capability.

¹⁴H. J. Lang, *Chem. Eng.* 55(6), 112-113 (1948).

¹⁵M. I. Lundin, *Economics of Power Generation - HRR*, ORNL CF-56-6-29 (June 4, 1956).

¹⁶J. D. Maloney, Jr., *Cost Estimates for Seven 200 Mw Turbine Plants for Operation With Nuclear Reactors at Various Steam Conditions*, ORNL-1387 (Dec. 23, 1952).

Fig. 20. HRR Turbogenerator-Station Payoff.

TABLE 11. TURBOGENERATOR-SYSTEM COSTS

Steam Temperature at Throttle (°F)	Cycle Efficiency (%)	Gross Electrical Generation (Mw)	Net Electrical Generation (Mw)	Installed Costs		Yearly Charge*
				Per Kilowatt	Total	
320	21.17	105.85	99.50	\$113.80	\$11,323,000	\$1,868,000
350	22.78	113.90	107.07	110.60	11,842,000	1,954,000
380	24.57	122.85	115.48	108.60	12,472,000	2,058,000
410	26.12	130.60	122.76	105.80	12,988,000	2,143,000
430	27.15	135.75	127.61	104.75	13,395,000	2,206,000

*Includes amortization, operation, and maintenance.

TABLE 12. HEAT-REJECTION-SYSTEM COSTS

Steam Temperature (°F)	Approximate Area Required* (ft ² × 10 ⁻⁶)	Installed Cost (\$)	Yearly Charge** (\$)
320	1.1609	1,358,300	287,700
350	1.0632	1,243,800	270,600
380	0.998	1,167,700	259,200
410	0.986	1,154,200	257,100
430	0.954	1,120,000	252,000

*Based on steam-to-air cycle; air at 95°F, 30°F approach.

**Includes amortization, power, and operating charges.

TABLE 13. YEARLY POWER REVENUE

(Dollars per Year at Load Factor Indicated)

Steam Temperature (°F)	Load Factor (%)				
	100	90	80	70	60
At 2 mills/kwhr					
320	\$1,743,000	\$1,569,000	\$1,395,000	\$1,220,000	\$1,046,000
350	1,876,000	1,688,000	1,501,000	1,313,000	1,126,000
380	2,023,000	1,820,000	1,619,000	1,416,000	1,214,000
410	2,151,000	1,936,000	1,721,000	1,506,000	1,290,000
430	2,236,000	2,012,000	1,789,000	1,565,000	1,341,000
At 2.5 mills/kwhr					
320	2,179,000	1,961,000	1,743,000	1,525,000	1,307,000
350	2,345,000	2,110,000	1,876,000	1,641,000	1,407,000
380	2,529,000	2,276,000	2,023,000	1,770,000	1,517,000
410	2,688,000	2,420,000	2,151,000	1,882,000	1,613,000
430	2,795,000	2,515,000	2,236,000	1,956,000	1,677,000
At 3 mills/kwhr					
320	2,615,000	2,353,000	2,092,000	1,830,000	1,569,000
350	2,814,000	2,532,000	2,251,000	1,970,000	1,688,000
380	3,035,000	2,731,000	2,428,000	2,124,000	1,821,000
410	3,226,000	2,904,000	2,581,000	2,258,000	1,936,000
430	3,354,000	3,018,000	2,683,000	2,348,000	2,012,000

TABLE 14. HIGH-PRESSURE-SYSTEM EQUIPMENT COSTS

Design pressure: 1400 psia

Design temperature: 600°F

Item	Description	Cost Basis	Cost	Number	Total Cost
Reactor vessel					
Pressure shell	10-ft-ID × 3.500-in.-thick sphere; type 347 SS clad, carbon-steel base.	60,000 lb at \$17.50 per lb, finished weight basis	\$1,050,000		
Thermal shield	9-ft-ID × 5-in.-thick sphere; SS clad on all exposed surfaces	57,000 lb at \$10.00 per lb	570,000		
Thimbles	6-in.-ID × $\frac{3}{4}$ -in.-thick cylinders; Zr-2 material; 80 ft long	3900 lb at \$50.00 per lb in the fabricated form	200,000		
					\$1,820,000
Heat exchangers	3280 straight-through $\frac{3}{8}$ -in.-OD × 18 BWG SS tubes; 8330 ft ²	Foster-Wheeler type of straight-through design from actual quotations	1,000,000	6	6,000,000
Circulating pumps	M-8000 Westinghouse canned-rotor centrifugal pump; 16,000 to 18,000 gpm	ORNL CF-56-4-126	250,000	2	500,000
Pressurizer					150,000
				Total	\$8,470,000

TABLE 15. STATION INVESTMENT

Reactor Plant		
Reactor-Plant Equipment		
High-pressure-equipment cost	\$ 8,470,000	
High-pressure-equipment piping	10,000,000	
High-pressure-equipment installation	5,000,000	
Low-pressure-equipment cost	5,670,000	
Low-pressure-equipment piping	4,900,000	
Low-pressure-equipment installation	2,440,000	
Instrumentation and controls	1,000,000	
Sampling equipment	100,000	
		\$ 37,580,000
Reactor-Plant Structure		
Containment vessels	\$ 900,000	
Cranes and handling facilities	750,000	
Cell ventilation and cooling	300,000	
Shielding	1,000,000	
Excavation and structural	500,000	
		\$ 3,450,000
Maintenance Facilities		5,000,000
Laboratories, Control and Administration Building		1,000,000
	Subtotal	\$ 47,030,000

TABLE 15 (continued)

Reactor Station		
Land Rights, Site Preparation, Improvements, etc.	\$ 1,000,000	
Waste Disposal	5,000,000	
Cooling Towers and Water System	200,000	
Chemical Processing Facilities	10,000,000	
	Subtotal	\$ 16,200,000
	Total Basic Costs	63,230,000
Contingencies $\pm$ (25%)		15,000,000
	Subtotal	78,230,000
Engineering and Design (10%)		7,800,000
Fees (3%)		2,350,000
	Total Reactor-Plant Cost	88,380,000
Turbogenerator Plant		
Turbogenerator	\$ 3,560,000	
Condenser, Cooling-Water System, Lubricating System, Piping, Auxiliaries	2,190,000	
Structures and Improvements	3,130,000	
Feedwater Equipment	2,000,000	
Accessory Electrical Equipment	1,250,000	
Miscellaneous	265,000	
Switchyard and Transmission	1,000,000	
	Subtotal	\$ 13,395,000
Total Station Investment		
Total Reactor Plant	\$88,380,000	
Turbogenerator Plant	13,395,000	
	Total	\$101,775,000

ACKNOWLEDGMENTS

This feasibility study could not have been prepared without the assistance and background information obtained from many members of the Laboratory staff. Where practical, this assistance is acknowledged in footnotes and in the Bibliography, although in several instances, where the authors reported on the work of entire groups, acknowledgement was not possible. Particular credit should be given for the helpful and guiding suggestions received during the course of the work from R. B. Briggs, W. R. Gall, J. A. Lane, A. M. Weinberg, and C. E. Winters, and from W. D. Reel in the writing of this report.

BIBLIOGRAPHY

- Arnold, E. D., A. T. Gresky, R. J. Klotzbach, and A. R. Irvine, ORNL-1761 (Jan. 27, 1955) (Classified).
- Aven, R. E., *Internal Recombination in the HRR*, ORNL CF-56-4-190 (April 26, 1956).
- Beall, S. E., and C. E. Winters, *The Homogeneous Reactor Experiment. A Chemical Engineering Pilot Plant*, ORNL CF-54-2-128 (May 4, 1954).
- Briggs, R. B., ORNL-1642 (May 4, 1954) (Classified).
- ETR Committee, *An Engineering Test Reactor for the MTR Site (A Preliminary Study)*, IDO-16228 (Oct. 1, 1954).
- Engineering Test Reactor Physics Progress Report* (Dec. 1, 1955), prepared for Kaiser Engineers by the General Electric Company.
- Falkenberry, H. L., and I. Spiewak, *Pulsafeeder and Purge Pump Failures on the HRT Mockup*, ORNL CF-56-3-173 (March 1, 1956).
- Foster Wheeler Corporation, *Progress Report - Heat Exchangers*, report No. 9 (Aug. 8, 1955).
- Gilfillan, D. R., ORNL CF-56-6-33 (June 5, 1956) (Classified).
- Gilfillan, D. R., and C. W. Nestor, Jr., ORNL CF-56-4-170 (April 26, 1956) (Classified).
- Gilfillan, D. R., and H. A. McLain, *HRR Converter Plates*, ORNL CF-56-6-28 (June 1, 1956).
- Griess, J. C., *et al.*, ORNL CF-56-4-138 (April 30, 1956) (Classified).
- Hughes, R. F., *The HRR Thimble Design Problem*, ORNL CF-56-6-175 (June 27, 1956).
- Kasten, P. R., *Control and Start-up of HRR*, ORNL CF-56-5-124 (May 18, 1956).
- Lansing, N. F., Sec. 4, "Aqueous Fuel Systems," Chap. 4.7, "Engineering Design Information," vol. 2, *Reactor Handbook*, Technical Information Service, USAEC, 1955.
- Lawrence, M. C., *Pressure Shell Stress Analysis HRR*, ORNL CF-56-5-188 (May 25, 1956).
- Lundin, M. I., *Canned Rotor Pumps for Homogeneous Research Reactor*, ORNL CF-56-4-126 (April 18, 1956).
- Lundin, M. I., *Economics of Power Generation - Homogeneous Research Reactor*, ORNL CF-56-6-29 (June 4, 1956).
- McLain, H. A., *HRR Heat Exchangers*, ORNL CF-56-5-143 (May 21, 1956).
- McNees, R. A., ORNL CF-56-6-166 (June 28, 1956) (Classified).
- McNees, R. A., ORNL CF-56-6-167 (June 28, 1956) (Classified).
- McNees, R. A., ORNL CF-56-6-168 (June 28, 1956) (Classified).

Maloney, J. D., Jr., *Cost Estimates for Seven 200 Mw Turbine Plants for Operation with Nuclear Reactors at Various Steam Conditions*, ORNL-1387 (Dec. 23, 1952).

Michelson, C., *HRR Piping*, ORNL CF-56-5-91 (May 14, 1956).

Michelson, C., *Maintaining High Pressure System of the HRR at Temperature After Dump*, ORNL CF-56-7-38 (July 10, 1956).

Michelson, C., *Revised Estimate of the High Pressure System Volume (HRR)*, communication, June 1, 1956.

Nestor, C. W., Jr., *Results of Preliminary Nuclear Calculations for a Homogeneous Research Reactor*, ORNL CF-56-4-75 (April 12, 1956).

Nestor, C. W., Jr., *Results of Preliminary Nuclear Calculations on Cylindrical Reactors*, ORNL CF-56-4-108 (April 16, 1956).

Ross, W. L., *Brief Status of the Development of Large Canned-Motor Pumps*, May 3, 1956 (personal communication).

Secoy, C. H., ORNL CF-56-5-114 (May 17, 1956) (Classified).

Segaser, C. L., *Zircaloy-to-Stainless Steel Transition Pipe Joint Design Study*, ORNL CF-55-12-72 (Dec. 5, 1955).

Segaser, C. L., and F. C. Zapp, ORNL CF-52-6-160 (Nov. 18, 1952) (Classified).

Thorium Breeder Reactor Design Study, unpublished ORNL Report (June 1955).

Westinghouse Electric Corp., *Hermetically Sealed Motor Pumps*, descriptive bulletin, 57-550 (Feb. 1955).

Wilson, J. C., and R. G. Berggren, *Effects of Neutron Irradiation in Steel*, paper presented at ASTM annual meeting, June 26, 1955.

Youngblood, E. L., *Feed Pump Mockup*, ORNL CF-55-2-52 (Feb. 9, 1955).

Zapp, F. C., *HRT Process Flowsheets*, ORNL CF-55-5-156, Revision II (March 6, 1956).

Zapp, F. C., and R. E. Aven, *HRT Design Data, Issue No. 2*, ORNL CF-54-8-53 (Aug. 10, 1954).