ANL-6451

ANL-6451

MASTER

325-19-62

Argonne National Laboratory

STATUS REPORT ON THE ARGONNE ADVANCED RESEARCH REACTOR

by

D. H. Lennox, E. W. Barts, R. V. Batch,
F. C. Beyer, G. L. Jorgensen, C. N. Kelber,
H. F. Reed, G. T. Rezek, G. S. Rosenberg,
H. C. Stevens, R. J. Weatherhead

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.

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, expressed 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, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

*Price \$2.75 . Available from the Office of Technical Services,
Department of Commerce, Washington 25, D.C.*

ANL-6451
Reactor Technology
(TID-4500, 16th Ed.,
Amended)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

STATUS REPORT ON THE ARGONNE
ADVANCED RESEARCH REACTOR

by

D. H. Lennox	C. N. Kelber
E. W. Barts	H. F. Reed
R. V. Batch	G. T. Rezek
F. C. Beyer	G. S. Rosenberg
G. L. Jorgensen	H. C. Stevens
R. J. Weatherhead	

November 1961

Operated by The University of Chicago
under
Contract W-31-109-eng-38

TABLE OF CONTENTS

	<u>Page</u>
I. INTRODUCTION.	11
A. Historical Background.	11
B. Present Status.	12
C. Summary	13
II. GENERAL DESCRIPTION OF THE REACTOR SYSTEM. . . .	15
A. Design Criteria	15
B. Facility and Site.	16
C. Reactor	16
III. REACTOR COMPONENT SPECIFICATIONS	21
A. Core Geometry and Composition.	21
B. Fuel Element.	24
1. Fuel Element Assembly.	24
2. Fuel Plates	24
C. Reflector	26
D. Core and Reflector Support Structure.	28
E. Reactor Vessel	28
F. Control and Instruments	31
1. Control Rod Drive and Poison Section	31
2. Nuclear Instruments	34
3. Process Instruments.	35
a. Pressure	35
b. Flow Rates	36
c. Level.	36
d. Temperature	36
e. Conductivity	36
4. Emergency Provisions	36
a. Power Failure.	36
b. Primary Coolant System Failures	36
c. Secondary Coolant System Failures	37
d. Shield and Experimental Facility Coolant System Failures	37
5. Radiation Monitoring.	37
G. Primary Coolant Systems	37
1. General Description	37
2. Discussion of Components	41

TABLE OF CONTENTS

	<u>Page</u>
H. Secondary Coolant System	42
1. Description	42
2. Water Treatment	42
3. Emergency Cooling	42
I. Auxiliary Systems	43
1. Demineralized Water Makeup System	43
2. Canal Cooling and Makeup System	43
3. Shield Cooling and Makeup System	43
4. Experimental Facilities Cooling and Cleanup System	43
5. Retention System	44
6. Boron Poison Injection System	44
IV. EXPERIMENTAL FACILITIES	45
A. Horizontal Beam Ports	45
1. Shutters	51
2. Maintenance	51
3. Safety	51
B. Beam Tubes	51
C. Experimental Service Facilities	52
D. Vertical Sample Irradiation Holes	53
V. BUILDING LAYOUT AND CONSTRUCTION	58
A. General	58
B. Plant Construction	62
1. Major Buildings	62
2. Service Structures and Facilities	62
3. Utilities	63
C. Shield Design Radiation Levels	63
D. Plant Containment	64
1. Design Criteria	64
2. Design Pressure and Shell Construction	65
3. Fuel Loading	66
VI. PLANT VENTILATION, HEATING, AND AIR CONDITIONING	69
A. Heating	69
1. Reactor Building	69
2. Stack Building	69
3. Equipment and Heat Exchange Building	69
4. Laboratory Building	70

TABLE OF CONTENTS

	<u>Page</u>
B. Ventilating	70
1. Reactor Building	70
2. Equipment and Heat Exchange Building	70
3. Stack Building	70
4. Laboratory Building	70
C. Air Conditioning	71
1. Reactor Building	71
2. Stack Building	71
3. Equipment and Heat Exchange Building	71
4. Laboratory Building	71
D. Exhaust Systems	71
1. Reactor Building	71
2. Equipment and Heat Exchange Building	72
3. Stack Building	72
VII. PHYSICS	73
A. Critical Experiments	73
B. Theory	75
1. Physical Configuration	75
2. Mode of Computations	76
a. Cross Sections	76
b. Computational Methods	78
3. Properties of the Clean Core	79
4. Long-term Burnup Properties	79
5. Geometric Effects	81
6. Reactivity Coefficients and Lifetime	82
7. Shutdown Recovery - Xenon Override	83
8. Storage Properties of the Fresh Core	84
9. Power Flattening	84
10. Stainless Steel Cores	86
11. Future Work	88
VIII. HEAT TRANSFER	89
A. Design Parameters	89
1. Fuel Heat Removal	89
2. Reflector Heat Removal	91
3. Shutdown Core Heat Removal	92
a. Normal Shutdown Requirements	92
b. Emergency Shutdown	92
c. After Heat Removal with No Forced Flow	92

TABLE OF CONTENTS

	<u>Page</u>
B. Critical Heat Flux	92
1. Forced Convection	92
2. Free Convection	94
3. Delay Time for Convective Cooling	95
C. Upflow Versus Downflow	96
D. Two-phase Flow Stability	96
E. Fuel Box Flow Tests	97
IX. SHIELDING	100
A. Reactor Radial Shield	100
1. Biological Shield	101
2. Thermal Shield	101
a. Shield Region Cooling Requirements	102
b. Construction Material	102
3. Reactor Vessel Thermal Shields	102
4. Beam Hole Shields	103
a. Beam Hole Shielding Plugs	106
b. Steel Liner for the Concrete	106
B. Shielding above the Core	106
C. Shielding below the Core	107
1. Thermal Shields below the Reactor Vessel	108
2. Concrete Shield below the Reactor Vessel	109
D. Control Rod Drive Room	110
E. Primary Coolant Pipe Tunnel	110
F. Spent Fuel	111
1. Spent-fuel Removal and Transfer Pool	111
2. Spent-fuel Transport Canal	112
3. Spent-fuel Storage and Handling Canal	113
4. Shielding and Radiation Dose	113
G. Heat Exchanger Building	114
1. Primary Coolant Heat Exchanger Cubicle	114
2. Secondary Coolant System Cubicle	115
3. Shield, Canal, and Experimental Cooling and Cleanup Systems	115
4. Primary Coolant Degassing, Cooling, and Cleanup System Cubicle	116
H. Spent-fuel Coffin	116
I. Irradiated Californium Samples	117

TABLE OF CONTENTS

	<u>Page</u>
APPENDICES	
A. Hydroelastic Problems in High-power -density Core Design	121
B. Two-phase Flow Stability in Parallel Heated Passages with Extensions to Current Concepts	126
C. Control Blade Temperature Calculations	137
D. Shielding Design Criteria, Parameters, and Calculations	142
ACKNOWLEDGMENTS	165

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	Plot Plan	17
2.	Reactor Elevation Showing Canal.	18
3.	Main Floor Plan.	19
4.	Reactor Elevation Showing Core Section.	22
5.	Core and Reflector Layout.	23
6.	Fuel Element.	25
7.	Reflector Details	27
8.	Reactor Pressure Vessel.	29
9.	Flow Diagram	38
10.	Horizontal Beam Facility Layout.	46
10a.	Reactor Plan Showing Horizontal and Vertical Facilities . . .	47
11.	Horizontal Beam and Through Tube Detail	49
12.	Reactor Elevation Showing Typical Beam Hole Service Facilities	50
13.	Beam Tube Service Equipment	54
14.	Vertical Thimble Details	55
15.	Hydraulic Rabbit Details	57
16.	Reactor Basement Floor Plan.	60
17.	Second Floor of Laboratory Building and Plan of Floor Over Heat Exchange Equipment	61
18.	Critical Experiment Core Region	73
19.	Idealized Core Layout for Physics Calculations	75
20.	Fuel Pattern for Flattened Power Distribution.	85
21.	Fuel Box Before Flow Tests.	97
22.	Inlet of Fuel Box After Flow Test	98
23.	Cf ²⁵² Transport Coffin.	118
B-1.	Pressure Drop vs Flow Rate in a Single Channel	126
B-2.	Pressure Drop vs Flow Rate in a Single Channel with Change in K	129

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
B-3.	Locations of Possible Roots of $S(\lambda)$ for One Channel Having $\partial\Delta P/\partial G < 0$	133
B-4.	Pressure Drop vs Flow Rate Showing Possible Location of "Limit of Operation" When Only One Channel Has $\partial\Delta P/\partial G < 0$	135
D-1.	Southwest Quadrant of AARR Critical Facility	160
D-2.	Measured and Calculated Dose Rates for AARR	161

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	Reactor Characteristics	20
2.	Core Dimensions and Compositions	21
3.	Reactor Vessel Design Data	30,31
4.	Control System Design Data	33,34
5.	Primary Cooling System Data	39,41
6.	Specifications for 100-Mw Cooling Tower	42
7.	Maximum Unperturbed Neutron Flux in Experimental Facilities	48
8.	Experimental Service Facilities	53
9.	Major Design Requirements for Vertical Facilities	56
10.	Building Layout Requirements	58,59
11.	Building Characteristics	62
12.	Service Structures and Facilities	62
13.	Allowable Dose Rates	64
14.	Design Pressures and Shell Construction	65
15.	Characteristics of Critical Assemblies	74
16.	Atomic Densities	75
17.	Group Structure	76
18.	Four-Group Cross Sections	77
19.	Reactivity Balance in Clean, Fresh Core	79
20.	Burnup Characteristics (Operation at 100 Mw).	80
21.	Reactivity Coefficients	82
22.	Setback Times	83
23.	Reactivity of Fractional Cores Stored in Water	84
24.	U^{235} and B^{10} Content of Fuels Used to Flatten Power	86
25.	Physics Constants for a Stainless Steel Core	87
26.	Heat Removal from the Fuel Region	90,91
27.	Heat Removal from the Reflector	91
28.	Burnout Heat Fluxes	93
29.	Overall Shield in the Direction of the Horizontal Through Holes	100

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
30.	Shipping Cask Shielding Data for a 60-Day Decay Core	117
31.	Radiation Dose Rates at Cf ²⁵² Cask Surface	119
D-1.	Radiation Dose Design Criteria	142
D-2.	Properties of Materials Used in the Shielding Calculations	146
D-3.	Core Gamma Source Intensities During Reactor Operation. .	147
D-4.	Core Gamma Source Intensities Two Hours After Shutdown.	147
D-5.	Reactor Midplane Gamma Fluxes on 5 Mw/Liter Core Operation (Primary Shield Arrangement)	149
D-6.	Conditions for the Production of Californium ²⁵² in the AARR	153
D-7.	Energy Release and Isotopic Concentration for Cf ²⁵² Production Sample	154
D-8.	Materials and Regions of the Cf ²⁵² Cask for the SNG Calculation	157
D-9.	Composition of AARR Critical Facility C-2 Prime	158
D-10.	Gamma Dose Rates in AARR Critical Facility (C-2 Prime) .	159
D-11.	Core 2 Operating Characteristics for Gamma Dosimetry Experiments	160
D-12.	Heat Generation Rates for Small Samples Along the AARR Horizontal Midplane at a Core Power Density of 2 Mw/Liter	163
D-13.	Core Gamma Ray Sources	164

STATUS REPORT ON THE ARGONNE ADVANCED RESEARCH REACTOR

I. INTRODUCTION

This is a status report on the interim design and development for an advanced research reactor (AARR) that will produce a maximum thermal neutron flux of 5×10^{15} neutrons/(cm²)(sec). The scope of the work done is limited to conceptual design studies supported by critical experiments plus heat transfer and hydraulic tests. Although this effort has been focused mainly on parameters fundamental to the technical feasibility of the concept, considerable time was spent to optimize the experimental utility of the reactor. As a result, this report also contains sections covering shielding, plant layout, fuel changing, maintenance, and auxiliary facilities. The information provided in these latter sections is not exhaustive, but it can serve as a guide for engineering design studies.

The AARR concept is for a light water-cooled, flux trap reactor with beryllium as the external reflector. At a power level of 100 Mw the expected maximum unperturbed thermal flux in the flux trap is 5×10^{15} neutrons/(cm²)(sec). The reactor is a dual-purpose research tool suitable for a multitude of beam experiments as well as for the production of transuranic isotopes and the irradiation of materials. The AARR will be of unique value to beam experiments where, for example, the study of elementary particles can be expedited and extended to new regions with a high-flux reactor. Fourteen beam holes penetrate the AARR beryllium reflector, some tangential and others radial, where the unperturbed neutron flux in certain regions will be as high as 2×10^{15} neutrons/(cm²)(sec). For the production of milligram quantities of transuranium isotopes or the acceleration of irradiation effects on samples, a flux trap is provided in the center of the fuel region.

A. Historical Background

The first ANL proposal (in March 1957) for a high-flux facility was the 250-Mw, D₂O-cooled-and-moderated reactor named "Mighty Mouse."¹ By virtue of a large D₂O reflector, this reactor provided maximum experimental volume for beam and irradiation facilities (49 vertical thimbles, 26 horizontal). Next, it became necessary to make an alternative, less costly design. After a number of parametric studies, a light-water-moderated-and-cooled core with a beryllium reflector was selected as the system offering the highest flux per dollar without an extensive development program.

¹L. E. Link et al., Terminal Report on the Mighty Mouse High-flux Reactor Project, ANL-5928 (Sept. 1959).

(Compared with the D₂O reactor design, a substantial cost reduction was effected at the expense of experimental volume.) A conceptual design for this second reactor, the AHFR, was completed in June 1959.² In September 1959 the project was reactivated to extend and refine the original AHFR design and also to carry out the research and development necessary to insure the performance of a high-flux, light water reactor. During the course of the following developmental program, substantial modifications and improvements to the initial concept evolved to give birth to the AARR - "The Argonne Advanced Research Reactor."

A major objective for the AARR program was to give reasonable assurance that an unperturbed thermal neutron flux of 5×10^{15} neutrons/(cm²)(sec) could be realized in the internal thermal column without compromising the reactor effectiveness as a facility for a variety of beam experiments. Since flux is proportional to power density, the first problem was to show that adequate heat transfer could be realized at the required power density. Preliminary estimates indicated that an average power density of 1.25 Mw/liter would be required, but the peak power density was not known accurately because of uncertainties in both the maximum-to-average power distribution over the fuel cycle as well as the effects of local perturbations due to control rods and inhomogeneities in the lattice structure. The need to define the maximum power density required was evident because the highest value reached in any operational reactor is approximately 1.8 Mw/liter (in the ETR). If the AARR should demand power densities greatly in excess of those attainable with current technology, a drastic revision of the basic design concept would be required.

To answer these questions, crucial to the feasibility of the conceptual design, critical experiments and burnout heat flux measurements were started. Also, a hydraulic loop was constructed to measure flow distribution within proposed fuel elements to establish hot channel factors and the velocity required to buckle the fuel plates.

B. Present Status

By May 1960, sufficient information had been obtained from the critical experiments³ and heat transfer measurements to make a reasonable design for a lattice that would be a close mockup for a final core configuration. Flow tests on preliminary fuel box models were in progress, but additional test data on hydraulic effects in the full core were needed to specify the structural requirements in detail. Construction plans for a

²L. E. Link et al., Argonne High-flux Research Reactor - AHFR, Conceptual Design Study, ANL-5983 (June 1959).

³J. W. L. DeVilliers, Editor, Critical Experiments for the Preliminary Design of the Argonne High Flux Reactor, ANL-6357 (June 1961).

second series of critical experiments that would be both a nuclear and mechanical duplicate of the AARR concept were started. Since the ZPR-VIII cell was already scheduled for use with another program upon completion of the exploratory measurements made for the AARR system, reactivation of the idle cell that formerly contained ZPR-V started. New piping and pumps were installed to complete the moderator transfer systems. Bids were let for the associated nuclear instrumentation and console wiring. Fuel and control element specifications were prepared consistent with the final evolution of the conceptual design for the AARR. (Initial critical experiments used surplus fuel and control elements from previous experiments which would not permit refined measurements of the local flux perturbations necessary to estimate design hot spot factors or control effectiveness and flux peaking factors.) In addition, more information was needed to select the most effective locations for beam holes.

In October 1960, budget uncertainties forced cancellation of further design and experimental work on the AARR, and effort on the project was transferred to other programs.

C. Summary

Throughout the AARR investigation, emphasis was placed on those basic physics and engineering experiments which would establish the feasibility of the conceptual design.

The nuclear properties of the concept were defined from exploratory critical experiments and theoretical calculations. The program included measurements of control rod worth, critical properties, temperature and void coefficients, macroscopic-flux distribution, fuel worth, and power-to-flux ratios.

Heat transfer measurements included tests for burnout under forced circulation and for pool boiling to simulate shutdown conditions. To conduct fluid flow tests on proposed fuel element designs, a loop was built to give velocity up to 50 ft/sec through one element. Tests were made of a parallel-plate box which was assembled by a series of keys at the plate edges. Thus, the channels were vented along approximately 75% of their length. This was done in an attempt to permit possible flow and pressure differentials between channels to equalize and prevent plate deformations at high flow velocities. A second advantage of the open design is the safety factor provided for natural-convection cooling in the event a fuel box should tip over while being stored in the canal.

To maximize the experimental utility of the reactor, studies were made of various layouts to provide maximum space for beam facilities and irradiation ports, as well as of effective ways to minimize downtime during

fuel-handling operations. To expedite fuel-changing operations, the reactor layout provides for both complete removal of the core as one unit directly to the storage canal or replacement of individual fuel boxes to save a partially used core or to alter the neutron spectrum at certain beam ports.

Numerous discussions were held with members of the Laboratory research divisions in order to coordinate conceptual design efforts with the needs of individual experimenters.

During the AARR experimental program and conceptual design study, it was expedient to investigate peripheral areas that indirectly influenced either the performance or utility of the reactor complex. In particular, information essential for a shield analysis also yielded data pertinent to such things as sample cooling requirements and waste disposal. Consequently, material was collected which goes beyond the scope of the usual conceptual design study.

Finally, all work that has now been done to explore the feasibility of the basic concept shows that the AARR performance objectives can be realized using the tools available under current reactor technology.

II. GENERAL DESCRIPTION OF THE REACTOR SYSTEM

A. Design Criteria

The design concept for the AARR was conditioned by the following ground rules:

1. The reactor should be a dual-purpose machine with emphasis on beam experiments and materials irradiation as well as on the production of transuranic isotopes.

2. For the production of isotopes or the irradiation of special samples, the unperturbed thermal flux at the target location should reach 5×10^{15} neutrons/(cm²)(sec). The flux magnitude for beam experiments, although it would have no rigid minimum threshold intensity, should be high enough to make significant experimental advances possible.

3. It should be possible to build and operate the reactor within a fairly short period of time, say four years. This, of course, ruled out an extensive development program and, at the same time, implied that the design would be strongly dependent upon conventional technology.

4. The cost of the reactor plus associated laboratories and facilities should not exceed 20 million dollars.

Later, as the conceptual design developed, it became apparent that the reactor should attempt to satisfy additional objectives. Since a high-flux facility represents a sizable investment, a long and useful lifetime is desirable. Consequently, the design should permit modifications to the reactor to cope with future experimental requirements or to profit from technological improvements. The reactor core structure should be completely removable so that at a later date it would be possible to use better fuel elements or change the lattice geometry. Also, replacement of the pressure vessel should be feasible for reasons of future flexibility, as well as the possible necessity for such action because of incomplete knowledge of long-term radiation-damage effects at the high flux levels contemplated. In view of the history of research reactor operation, major components of the system should be sized to permit a 50% increase in the reactor design power level. Finally, conventional technology is pushed to the limit to obtain the high power densities consistent with the required neutron flux levels. As a result, a fuel meltdown accident is conceivable and the primary system must be designed to permit rapid decontamination, and abundant facilities for waste disposal should be provided.

B. Facility and Site

The proposed location (Fig. 1) for AARR is near Building 330 (CP-5) to permit convenient interchange of experimental equipment and common usage of some facilities. The site layout gives a clear path to two 400-meter beams for time-of-flight experiments.

The reactor is enclosed by a 120-ft-diameter steel dome. Adjacent buildings contain laboratory and office space, an exhaust stack with filtering equipment, cooling water and waste treatment equipment, and primary and secondary heat exchange equipment. Retention basins and water storage tanks complete the complex.

Because space within the containment shell is expensive, only items of experimental utility or vital to operation are inside. The containment building will house the reactor, the control room, and a portion of the fuel-handling canal (see Fig. 2). To make efficient use of the floor area, the reactor is located off-center (see Fig. 3). Beam holes nearest the building wall are designed for the more compact experiments, whereas on the opposite face of the reactor, extra clearance is provided for experiments that may require additional room. The canal, which leads to the fuel-storage area, passes under the containment shell to the equipment building, where final transfer of samples or fuel is made. The junction between the two buildings is sealed by an underwater gate in the canal. Connecting pipes to the reactor run in a trench below the canal to the primary and secondary heat exchange components located in the equipment building. Inside the containment building the canal terminates over the top of the reactor, so that fuel can be lifted directly from the reactor after the primary system is depressurized.

Access to the containment building is through a personnel air lock from the laboratory and office building, or through an equipment air lock leading to the equipment and storage building. Additional openings provide for an emergency air lock, a large equipment hatch, and ventilating and exhaust ducts.

C. Reactor

The AARR is a 100-Mw research reactor with an internal, light water thermal column for isotope production, and an external beryllium reflector for beam tubes. The flux is peaked in both regions to give 5×10^{15} neutrons/(cm²)(sec) in the flux trap and 2×10^{15} neutrons/(cm²)(sec) in the outer reflector. The active lattice consists of modified MTR-type fuel elements cooled by light water. A high metal-to-water ratio (one-to-one) is employed to enhance the leakage into the flux trap. The fuel density throughout the 77-liter core is uniform, and burnable poison in the form of boron is added to the fuel to decrease the number of movable rods needed for reactivity compensation.

FIG. 1
PLOT PLAN

FIG. 2
REACTOR ELEVATION SHOWING CANAL

FIG. 3
MAIN FLOOR PLAN

Four cadmium rods within the fuel region are used for shim purposes, while four other cadmium blades at the fuel-beryllium interface are for shutdown safety. The fuel cycle is twelve days with an additional one day for recharging. Characteristics of the reactor are summarized in Table 1.

Table 1

REACTOR CHARACTERISTICS

Reactor name	Argonne Advanced Research Reactor
Type	Research
Location	Argonne, Illinois
Operator	Argonne National Laboratory
References	ANL-6357 ³
Reactor heat output	100 Mw
Coolant-moderator	H ₂ O
Fuel	U ₃ O ₈ (93.5% enriched)
Reflector	Beryllium radial, H ₂ O axial
Volume of core	77.7 liters
Maximum thermal neutron flux	5×10^{15} neutrons/(cm ²)(sec), (central flux trap)
Average fuel cycle	12 days
Maximum power density	4.7 Mw/liter
Coolant outlet temperature	80°C (180°F)
Coolant inlet pressure	20.4 atm (300 psig)

³ See page 12

III. REACTOR COMPONENT SPECIFICATIONS

A. Core Geometry and Composition

The overall composition of the core can be rather exactly specified from critical experiments on the lattice plus theoretical calculations. Also, the total heat transfer area necessary to realize the maximum power density required is based on critical experiments and heat transfer measurements. Probably the least defined area of the design concerns mechanical details of the core, because only preliminary flow tests were made on one proposed design for a fuel element. Initially, upflow through the core was considered; but after a review of the loss-of-coolant-flow type of accident, it was found that the thermal buoyancy advantages of upflow were negligible. Consequently, downflow was selected for its attendant mechanical features, such as the removal of a top support grid, safety rod scram assist, and the elimination of a restraint on axial expansion. Figure 4 shows an elevation of the reactor core and pressure vessel; Fig. 5 gives the core layout.

The dimensions and compositions of the reactor fuel region and reflector are listed in Table 2.

Table 2

CORE DIMENSIONS AND COMPOSITIONS

1. Fuel Region		
Active cross-sectional dimensions		49.84 cm sq (19.62 in. sq)
Active height		44.45 cm (17.5 in.)
No. fuel boxes		48
Fuel plates per box		30
Plate dimensions		49.53 x 5.91 x 0.10 cm (19.5 x 2.328 x 0.04 in.)
Total core composition		
U ²³⁵ (as 93.5% enriched U ₃ O ₈)		6.5 kg (14.33 lb)
Aluminum		101 kg (222.7 lb)
B ¹⁰ (contained in natural boron)		7.2 gm (0.0159 lb)
H ₂ O		38.9 kg (85.77 lb)
2. Internal Thermal Column		
Cross-sectional dimensions		12.5 cm sq (4.921 in. sq)
Height		44.45 cm (17.5 in.)
3. Beryllium Radial Reflector		
Cross-sectional dimensions		
Outside		114 cm sq (44.88 in. sq)
Inside		53 cm sq (20.866 in. sq)
Height		45.7 cm (17.992 in.)
Composition		
Volume (Be + H ₂ O)		453 liters (15.914 ft ³)
Beryllium		388 liters (13.630 ft ³)
Water		43 liters (1.511 ft ³)

FIG. 4
REACTOR ELEVATION SHOWING CORE SECTION

FIG. 5
CORE & REFLECTOR LAYOUT

B. Fuel Element

1. Fuel Element Assembly

The fuel element assembly as shown in Fig. 6 consists of a group of 30 fuel plates fastened together at the edges by means of tapered locking keys and at the bottom by a nozzle which positions the bottom of the element in a lower support grid. The purpose of the tapered locking keys is to provide a discontinuous way to lock the individual fuel plates into a rigid cluster in order to leave openings between the plates to relieve possible pressure differentials between channels. The distance between plates is maintained by spacers.

2. Fuel Plates

The fuel plate is Al-U₃O₈ cermet clad with aluminum. The tentative specifications for the fuel plates are as follows:

a. The fuel core must be a homogeneous aluminum-uranium oxide dispersion whose composition is such that each core shall contain 4.52 ± 0.09 gm of U²³⁵. Homogeneity of the uranium core shall not vary by more than 3% from the average composition throughout a given core.

b. The uranium oxide shall be in the form of U₃O₈ enriched in U²³⁵ to a minimum of 90%. The uranium oxide powder shall be minus 325 mesh and relatively free from agglomerates which may give rise to fragmentation and stringing during fabrication.

c. The aluminum used for the matrix shall be a powder whose analysis conforms to the Aluminum Alloy Designation System No. 1100. The nuclear poisons shall be limited to a maximum corresponding to 0.3 ppm cadmium and 3.0 ppm total rare earth elements.

Boron which normally occurs in the aluminum alloy shall be taken into account to determine the total amount of burnable poison to be added (see specification "g").

d. The cladding shall be metallurgically bonded throughout all mating interfaces. The bond between cladding and uranium core components shall be complete to a minimum of 99.5% of the interfacial area. No defect greater than $\frac{1}{32}$ in. in any dimension shall be present in the bond; and no two such defects shall be within $\frac{1}{2}$ in. of each other. The bonds shall be free of contamination which might lower the corrosion resistance of the aluminum and be free of uranium. Any inclusions or voids found in the bonds shall be cause for rejection.

Although the best current ultrasonic testing methods can detect flaws of $\frac{1}{16}$ -in. diameter, it is hoped that these techniques can be refined.

NOTE: ALL DIMENSIONS SHOWN IN INCHES

FIG. 6
FUEL ELEMENT

e. The cladding shall be aluminum whose analysis conforms to Aluminum Alloy Designation System No. 6061. The aluminum used in the fabrication of plates shall be free of cracks, checks, seams, laminations, striations, and loose or embedded impurities. The nuclear poison materials (boron, cadmium, and the rare earths) shall be limited to the following maximum quantities:

Boron	0.5 ppm
Cadmium	0.3 ppm
Total rare earths	3.0 ppm

If the boron content in the cladding exceeds specifications, the excess can be subtracted from burnable poison requirements.

f. The finished fuel plate must be free from surface imperfections (such as laps, seams, metallic, and nonmetallic inclusions) and from radioactive surface contamination. Dents, scratches, or pits deeper than 0.0005 in. will be cause for rejection if located in the cladding directly over the core.

g. Each fuel plate shall contain 0.0050 ± 0.001 gm B¹⁰ in a homogeneous mixture of natural boron with the uranium oxide and aluminum. Homogeneity of the boron in the mixture shall not vary more than $\pm 3\%$ of the average composition per unit length throughout each plate.

C. Reflector

The inner region next to the fuel is a single belt of 3 x 3 in. (7.62 cm sq) beryllium blocks slightly longer than the fuel. No beam holes enter this region and, as a result, the beryllium can be easily replaced when radiation damage makes this necessary. Also, other materials (including fuel) may be substituted for a limited number of these square cross-section pieces to alter the neutron spectrum at the appropriate beam port.

The outer reflector region is made from larger beryllium blocks, to reduce the machining operations necessary for beam holes. The blocks are cut in three horizontal layers corresponding to each of the elevations for beam tubes. Two through tubes (HT1 and HT2) are in the bottom layer. The middle layer of beryllium contains beam tubes that lie in the core midplane (HB1, HB2, HB6, HB7, HB8, HB9, and HB10). Nine vertical sample holes penetrate all three layers. The reflector details are given in Fig. 7.

To conserve reactivity, the water content of the beryllium reflector is less than 10%. Nonuniform distribution of coolant passages is necessary to satisfy this criterion.

The inner reflector pieces, where maximum heat generation occurs, require excess water for adequate cooling - four $\frac{5}{16}$ -in. (0.794-cm) holes are required in each 3 x 3 in. (7.62 cm sq) block. The number of cooling passages is decreased in the outer region of the reflector to bring the average water content down to the acceptable limit specified. These holes in the beryllium are the primary orifices used to give proper flow distribution between fuel and reflector, because a permanent restriction in the grid plate would prevent substitution of materials with greater rates of heat generation.

D. Core and Reflector Support Structure

The core and reflector support structure may be seen in Fig. 4. The fuel elements rest in an aluminum grid plate which is 3 in. thick and has holes bored in it to receive the bottom adapter section of the elements. The grid plate is supported by four aluminum plates, 6 in. wide by $\frac{1}{2}$ in. thick (15.24 cm wide by 1.27 cm thick). The support plates are mechanically fastened to the underside of a 5-in. (12.70-cm) thick stainless steel reflector support plate. The reflector support plate has numerous holes bored in it to allow passage of water for cooling the beryllium reflector. The reflector support plate is bolted to a similarly shaped ring on the support shroud which, in turn, locates the core and reflector in the pressure vessel. All structure for core and reflector support can be replaced or modified if required.

E. Reactor Vessel

Details of the reactor pressure vessel are shown in Fig. 8.

The top section of the reactor vessel is a right cylinder of diameter sufficient to permit free removal of the core, reflector, and grid plate. The door in the top cover is used for routine fuel changes so that vertical thimbles and experimental ports need not be disturbed until major maintenance requires complete cover removal. At midsection, the vessel is spherical; the diameter is increased to eliminate extensive internal thermal shielding and to give a circumference that leaves adequate ligament area between beam hole penetrations. The bottom section is again a right cylinder, but of smaller diameter than the top, because only access to the control components is necessary from this direction.

The specifications for the pressure vessel include allowances to permit a future increase in reactor power from 100 to 150 Mw. A vessel lifetime, based on radiation damage estimates, of 20 yr is anticipated. Design data for the vessel are given in Table 3.

Table 3

REACTOR VESSEL DESIGN DATA

Design Pressure	450 psi
Temperature	
Design temperature	460°F
Bulk water temperature	205°F
Overall Dimensions	
Total height	16 $\frac{1}{2}$ ft
Maximum diameter, including mounting lugs	9 $\frac{1}{4}$ ft
Top cylinder	
ID	54 in.
Thickness	1 in.
Length, including knuckle	50 in.
Spherical mid section	
ID	96 in.
Thickness	$\frac{3}{4}$ in.
Bottom cylinder	
ID	42 in.
Thickness	$\frac{3}{4}$ in.
Length to sphere	53 in.
Top Closures	
Major closure - circular flat head	
ID	54 in.
Thickness	6 in.
Minor closure - rectangular	24 x 27 in.
Bottom head - spherically dished	
Depth	9 $\frac{1}{2}$ in.
Support	lugs
Nozzles and Post	
Coolant inlet, ID	24 in.
Coolant outlet, ID	24 in.
Vertical Be thimbles	
No.	6
ID	2 in.
Vertical H ₂ O thimbles	
No.	11
ID	3 in.
Vertical central thimble	
No.	1
ID	9 in.
Horizontal through tubes	
No.	2
Entrance ID	9 in.
Exit ID	5 in.

Table 3 (Cont'd.)

Nozzles and Post (Cont'd.)

Control rod ports

No.	12
ID	2 in.

Hydraulic rabbits

No.	2
ID	2 in.

Boron injection port

No.	1
ID	3 in.

Internal Thermal Shields

small, local

Other Internal Parts

Core Support

bottom cylinder
extension

Materials of Construction

Vessel

Type 304 stain-
less steel

Cladding

None

Size

Weight	~25,000 lb
Volume	~365 ft ³

Construction Standards

ASME Pressure Vessel Code where applicable
and adequate for performance criteria

Special Design Features

- a. Vessel can be replaced
- b. Only local thermal shielding
- c. Special closure for loading core
- d. 34 experimental and sample penetrations.

F. Control and Instruments.1. Control Rod Drive and Poison Section

For control regulation and shutdown, twelve poison blades are actuated by individual drive mechanisms located in the subreactor room. Four blades are located in the core radially to the axis, while the remaining eight surround the core at the beryllium interface.

Each control blade is 102 cm long and 0.51 cm thick. The width of the radial blades are 12.7 cm and the blades along the outer perimeter are either 19 cm or 29.2 cm wide. The top half of each blade contains the poison section consisting of 0.176 cm of cadmium sandwiched between 0.165 cm of aluminum. The remaining portion of the blade is of aluminum and serves as a follower. Each blade operates in an aluminum sheath which extends 51 cm above the core. The sheath provides support for the blade throughout its 51-cm travel, while also providing the channels for blade coolant. Blades are withdrawn from the core in an upward direction.

The control blade is connected to the drive mechanism by means of an extension rod that can be connected to allow the control blades to remain with the spent fuel upon core removal. The extension rods pass through the bottom reactor flange to the subreactor room. A shutoff valve, integral with the extension rod, prevents loss of reactor water should removal of the labyrinth-type pressure seal become necessary. Surrounding the extension rod within the reactor vessel is a guide tube that furnishes support and shields the rod from dynamic loading by the stream of core coolant.

The drive mechanism (located below the reactor in the subpile room) is a combination air-cylinder-motor-driven lead screw assembly. The control blade and extension rod are connected to the piston of the continually pressurized air cylinder. The motorized lead screw is mounted to the cylinder and actuates a nut which moves parallel to and in the same direction as the cylinder piston. A bracket integral with the nut rests against the top of the piston and prevents upward piston motion. As the lead screw turns, raising the nut, the piston and control blade will move up, following the nut. Similarly, downward motion of the nut will force the piston and blade downward. For rapid insertion, an accumulator is connected to the rod end of the cylinder through a solenoid shutoff valve. Interruption of the solenoid current will open the valve and flood the cylinder with high-pressure air. Insertion times can be regulated by varying accumulator pressure. A hydraulic dashpot at the base of the cylinder decelerates the blades at the "full in" position. Since the poison section is withdrawn above the core, gravity plus coolant downflow makes the rod fail-safe.

Indication of blade position is achieved through a synchrotransmitter and receiver. A gear coupled directly to the lead screw drives the transmitter through a set of reduction gears. Limit switches restricting blade movement are also actuated directly from the lead screw.

Table 4 is a summary of design data for the control system.

Table 4

CONTROL SYSTEM DESIGN DATA

Control Blades

Number	12
Configuration	Rectangular
Overall dimensions	4 blades, 0.51 x 12.7 x 102 cm 4 blades, 0.51 x 19 x 102 cm 4 blades, 0.51 x 29.2 x 102 cm
Absorber material	0.176 cm of cadmium clad with 0.165 cm of aluminum
Absorber length	51 cm
Follower material	Aluminum
Follower length	51 cm
Weight of blades	2.5 kg
Coolant flow channel between blade and sheath	0.102 cm
Pressure drop across blades	4.76 atm
Maximum blade temperature (in core)	108°C
Maximum blade temperature (out core)	121°C
Coolant velocity through control channel (max.)	12.9 meters/sec
Coolant temperature rise (max.)	19°C

Drive Mechanism

Number of Drives (max.)	12
Type	Pneumatically actuated scram and lift, with lead screw driven positioning and re- latching. Integral hydraulic dashpot

Table 4 (Cont'd.)

Components

Cylinder	5.08-cm, double-acting air cylinder, with 2:1 oversize rod
Drive motor	$\frac{1}{4}$ -hp, 3-phase, 60-cycle gear-head motor arranged for vertical mounting. Motor equipped with uni-brake.
Lead screw	$1\frac{1}{4}$ - 5 P Acme screw - steel
Nut	$1\frac{1}{4}$ - 5 P Acme nut - bronze
Dashpot	Hydraulic, tapered plunger type made integral with cylinder
Rod extension material	17 - 4 PH stainless steel (H-1100 temper)-chrome plated
Rod seal	Labyrinth-type pressure breakdown ring seals
Accumulator	Suitable capacity to scram cylinder in allotted time
Compressor	To maintain cylinder pressure which will be under 7 atm
Indication	Synchrotransmitter and receiver
Miscellaneous components	Micro switches, solenoid valves, pressure switches, etc.
Regulating speed	Not fixed at this time.
Scram time	Not fixed at this time.

2. Nuclear Instruments

For operation of a reactor with the characteristics of the AARR, there is a considerable incentive to develop a computer-type control system that would minimize downtime and prolong the effective core lifetime. Such a system might, for example, "remember" the reactivity history and control rod program for the core. After a power transient, or set-back, the computer would predict the most effective power recovery schedule and the

required rod motion to accomplish this without exceeding limitations of local power density. At this time, it is not possible to give specifications for a supplementary automatic system; however, the basic requirements for conventional instruments is listed.

The measurement of neutron flux will be accomplished through the use of 15 detection channels.

- a. Channels I, II, and III - Log count rate-period
 Detector - Fission chamber
 Range - 0.167 to 1.67×10^5 counts/sec. Period from minus 30 sec to infinity to plus 3 sec
- b. Channels IV, V, and VI - Log flux-period
 Detector - Compensated B_{10} ion chamber
 Range - 1.5×10^{-4} to 5×10^{-11} amp
 Period from minus 30 sec to infinity to plus 3 sec
- c. Channels VII and VIII - Linear flux
 Detector - Compensated B^{10} ion chamber
 Range - 10^{-3} to 10^{-12} amp
- d. Channels IX, X, and XI - High-flux safety trips
 Detector - Uncompensated B^{10} ion chamber
 Range - 10^{-3} to 10^{-6} amp
- e. Channels XII, XIII, XIV, XV, - Control as required.

Four instrument thimbles having an inside diameter of 18 cm will be required for the neutron detectors listed above. These instrument thimbles should be located at 90-degree spacings around the core and should see a thermal neutron flux of 10^{10} neutrons/(cm²)(sec) at the detectors for a reactor power of 100 Mw. An adjustable locating device should be incorporated with the thimbles for positioning the neutron detectors. A minimum flux level of 2 neutrons/(cm²)(sec) will be required with a new core when all rods are in.

3. Process Instruments

Some general guidelines established for measurement of plant factors are as follows:

a. Pressure

Pressure will be measured with bourdon tube actuated beam balance transmitters having a 1.0- to 5.0-ma, dc output.

b. Flow Rates

Flow rates will be measured with differential pressure cell actuated beam balance transmitters having a 1.0- to 5.0-ma, dc output connected across flow nozzles or orifices.

c. Level

Liquid level will be measured with differential pressure cell actuated beam balance transmitters having a 1.0- to 5.0-ma, dc output.

d. Temperature

Temperatures will be measured with iron-constantan and chromel-alumel thermocouples, and temperature resistance bulbs.

e. Conductivity

Water conductivity will be measured with temperature-compensated conductivity cells. Continuous-reading pH meters are required to monitor the water in the primary system, shield, and canal.

4. Emergency Provisions

a. Power Failure

In the event of a normal power failure, an automatic reactor shutdown is effected, and a diesel generator starts to supply auxiliary power. The diesel generator requires about 10 sec to start and come to the proper speed. During this 10-sec interval, uninterrupted power is supplied in: (a) emergency and shutdown pumps; (b) the experimental facility coolant system pump; (c) certain process nuclear and experimental facility instrumentation; and (d) miscellaneous necessary loads. This intermediate source of uninterrupted electrical power consists of a motor-generator-alternator driven from a station battery which can supply only a limited number of kw-sec of electricity. These emergency loads must, therefore, be switched over to the diesel generator power supply as soon as possible. Other plant loads (not requiring uninterrupted electric power) that are essential to complete reactor shutdown will also be connected to the diesel generator supply.

b. Primary Coolant System Failures

A primary coolant system rupture causes immediate reactor shutdown. Process water from the storage tank is available for core cooling or the canal water may be used. If the flow rate of the primary system decreases, due to the mechanical failure of one or both of the coolant pumps,

it may be desirable to program the reactor power to maintain a normal value of reactor outlet temperature, in order to complete reactor shutdown and maintain over-ride capability. The controls necessary to handle this type of emergency situation, however, require study.

c. Secondary Coolant System Failures

A secondary coolant system rupture also causes an automatic reactor shutdown. Process water from the storage tank or canal can be diverted to the primary heat exchangers to provide a temporary heat sink. If the flow rate of the secondary system decreases due to the mechanical failure of one or both of the coolant pumps, again it may be possible to program the reactor power to maintain a normal value of reactor outlet temperature. In the event of a loss of the makeup flow to the secondary system, the reactor is shut down. Process water from the storage tank can be diverted into the secondary system to provide the necessary temporary makeup water.

d. Shield and Experimental Facility Coolant System Failures

Emergency provisions in the event of system rupture or mechanical failure of the coolant system pumps will be made, but additional study is required to define the individual requirements.

5. Radiation Monitoring

Conventional health physics instruments are required to monitor for radiation and particulate matter within the reactor and equipment buildings.

The primary and secondary reactor coolant systems will require detection systems for activity from fuel element or sample failures. No specifications have been established for these instruments.

G. Primary Coolant Systems

1. General Description

The main coolant system is designed for normal operation at 100 Mw; however, 150-Mw operation is possible by proper choice of temperature differentials.

A flow rate of approximately 20,000 gpm is required to cool the reactor core, reflector, and internal structure. Deionized water enters at the upper cylindrical section of the pressure vessel and flows downward through the core and reflector. The water then leaves the pressure vessel at the lower cylindrical section of the pressure vessel after making one pass through the core and reflector (see the AARR Flow Diagram shown in Fig. 9).

FIG. 9
FLOW DIAGRAM

Only the pressure vessel and a portion of the piping are located in the containment building. All other components are located in the heat exchange building.

In the heat exchange building, hot water from the reactor first passes through a trap to collect any sizable particles that may have entered the coolant stream. From the trap, the water then flows through the heat exchangers and to the pumps. The cold water then returns directly to the pressure vessel.

Design conditions for the primary coolant system that can be specified at this time are tabulated in Table 5.

Table 5

PRIMARY COOLING SYSTEM DATA

Number of loops	1
Flow rate	20000 gpm (75,700 liters/min)
Coolant operating conditions	
Average inlet temperature (reactor)	135°F (57°C)
Average outlet temperature (reactor)	
for operation at	
100 Mw	176°F (80°C)
150 Mw	196.5°F (91°C)
Maximum velocity (through core)	40 ft/sec (12.2 m/sec)
System pressure drop	105 psi (7.14 atm)
System pressure, design	465 psia (31.6 atm)
System temperature, design	459°F (237°C)
System pressure, cold hydrostatic test	675 psig (45.9 atm)
Loop components	
Piping	
OD and wall thickness	24-in. OD x 0.500-in.
approximate length	205 ft
material of construction	304 SS
Valves	
type	ball
number of isolation valves	12 - 18 in.

Table 5 (Cont'd.)

number of throttle valves	8 - 12 in.
drive mechanisms	
isolation valves	hand operated
throttle valves	motor driven
Main coolant pumps	
type and number	2 vertical turbine
flow, each	10000 gpm
temperature	135°F
design head	730 psig
brake horsepower at full flow	865 hp
efficiency at rated power	87.5%
time limit that motor-pump unit can operate with primary coolant below 359°F	continuously
construction materials	Type 304 stainless steel
electrical input	4140 v, AC, 3-ph, 60 cps
Heat exchangers	
number and type	4 two-pass, vertical
dimensions	28 ft long x 60 in.
surface area	32,000 sq ft
overall heat transfer coefficient	300 Btu/(ft ²)(hr)(°F)
construction	2280 tubes per unit 5/8 in. - 18 BWG tubing on 7/8 in. - square pitch
Coolant inventory	
system liquid volume	2015 ft ³
reactor vessel volume without internals and core	435 ft ³
pipes, fittings, and valves, liquid volume	1042 ft ³
pumps (total for two), liquid volume	38 ft ³
heat exchanger (total for 4), liquid volume	500 ft ³
coolant cycle time	45 sec

Table 5 (Cont'd.)

Convection head

elevation of heat exchangers above core centerline	3 ft
---	------

2. Discussion of Components

The two primary circulation pumps that operate in parallel are of the vertical turbine type to minimize floor space requirements. For the same reason, the four main heat exchangers, in parallel, are vertical. This position is expected to also help maintenance.

All piping used in the primary system is specified ASTM SA-312, Grade TP 304. The fittings are to be ASTM SA-182, Grade F 304. Ball-type valves are favored throughout the primary system.

A separate tank connected to the main coolant inlet line of the reactor is required to pressurize the system and also to function as a surge tank. The best method for pressurizing the system to guarantee sub-cooling during normal and emergency situations has not been defined.

When the reactor is in operation, a constant volume of water is bled off to the degasifier. From the degasifier the water enters a heat exchanger and is then pumped through filters and an ion exchange column before returning to the inlet of the main circulating pumps. All makeup water goes through the degasifier, filters, and ion exchange column before entering the primary system. Thus, the purity of the primary system water is maintained by recirculating water constantly through the degasifier filters and ion exchange column. The conductivity and pH control of the primary water will be of critical concern because the scale ΔT in the core must be limited to around 50°F. A pH of 5.5 with megohm-quality water is indicated.

To provide for cooling the reactor core during times when the main coolant pumps are inoperative, a DC motor-driven pump will circulate approximately 1000 gpm of primary water. Power will be furnished to the DC motor by batteries. These batteries will be kept fully charged by a diesel-generator set which can also be used to supply power for other equipment under emergency conditions.

The use of a flywheel on the main pumps and an auxiliary motor on the main pump motor shaft has also been considered as means of providing power for emergency cooling. Each of these systems has its own unique advantages, and a final design may incorporate more than one means of supplying power for emergency cooling.

H. Secondary Coolant System

1. Description

The secondary coolant system is designed to dissipate to the atmosphere by means of cooling towers the 100 Mw (or 150 Mw) of heat that the heat exchangers have removed from the primary coolant. The secondary water flow is from the heat exchangers to the cooling tower, then to the water-treatment building for purification, and then to the pumps for recirculation. Makeup water is added in the water-treatment building after purification. This system also provides cooling water for auxiliary systems. Two secondary pumps are used in parallel operation. Here, as for the primary system, vertical turbine pumps are specified.

The cooling tower will consist of a number of cells operating in parallel. They will have counter flow induced by adjustable pitch fans. Upward spray of the water is used with a tower height of 40-42 ft. The cooling tower specifications are given in Table 6.

Table 6

SPECIFICATIONS FOR 100-Mw COOLING TOWER

Design conditions:

Wet bulb temperature	80°F
On-tower water temperature	120°F
Off-tower water temperature	90°F
Total Water Flow	22,900 gpm
Water Concentration	2.5 gal/(min)(ft ²)
Tower area	9,150 ft ²
Fan hp	380 hp
Makeup	800 gpm

2. Water Treatment

Conventional methods of water treatment will be used. The water will be continuously monitored for radioactivity, checked for hardness, pH, and algae.

3. Emergency Cooling

A battery-powered, DC motor-driven pump will circulate water at a rate of 1700 gpm through the heat exchangers. The battery set which furnished power for the primary system emergency pump will also furnish

power for the secondary system emergency pump. Tower makeup water will be supplied from either an emergency pump system or from the plant fire system.

I. Auxiliary Systems

1. Demineralized Water Makeup System

The mineralized water makeup system will be designed to satisfy all pure water requirements of the plant. The ion exchange equipment will be located in the heat exchange building. Pure water is stored in an elevated tank which has a capacity of 20,000 gal. The elevation of this tank is approximately 100 ft above the reactor core, and thus can be used to supply cooling water if normal methods of emergency cooling fail.

2. Canal Cooling and Makeup System

The canal cooling and makeup system will provide for continuous purification and cooling by recirculation of water through a heat exchanger, resin columns, and filters. Inlet water is taken from the spent core section of the canal to assist convection cooling of the fuel. An automatic level control will maintain a constant depth of water in the canal by adding water as required to replace losses from evaporation or leaks. In an emergency, large amounts of water can be taken from the service mains and added to the canal. The canal floor, however, is designed to withstand moderate impact loads from dropped coffins in order to prevent gross leakage. A scum gutter around the top inside edge of the canal will limit the maximum water depth.

3. Shield Cooling and Makeup System

The shield cooling and makeup system will continuously cool and purify the water which is in the annulus between the pressure vessel and concrete shielding. The upper shielding blocks do not provide a water-tight seal between the canal and the water shield; hence, there will be a very small flow of water between them. This can be minimized by adjusting the pressure differentials. A flow in either direction will not be harmful, but it will require some water makeup to one system or the other. The water in this annulus will also cool the thermal shields.

4. Experimental Facilities Cooling and Cleanup System

Each beam tube facility will be provided with a source of pure water for cooling purposes. The supply and return lines for each beam tube facility are connected to a common heat exchanger and ion exchange column. If a particular beam tube experiment is hazardous and the possibility of cross-contamination exists, then an auxiliary cooling and cleanup system must be supplied with the experiment.

5. Retention System

All radioactive or suspect liquid waste from the primary system or any of the auxiliary systems will be piped to the retention basins. Four retention basins, each of 125,000-gal capacity, will be provided. The liquid waste will be kept in these basins for cleanup or decay until safe disposal is possible.

6. Boron Poison Injection System

For backup safety or possible use in emergency situations, an injection system to put boric acid solution into the coolant is visualized. Sufficient quantities of the solution would be required eventually to poison all of the water in the primary system. The point of injection would be in the core region for fast initial action. The boron poison-injection system should operate on automatic control and from a manual signal.

IV. EXPERIMENTAL FACILITIES

There are 35 experimental facilities, consisting of 12 horizontal beam tubes, two through tubes, and 21 vertical thimbles. Additional sample irradiation volumes can be had in a region of high flux by modification of the inner reflector blocks.

A. Horizontal Beam Ports

Each of the 12 horizontal beam holes penetrates the reactor shield, reactor pressure vessel, and then terminates in the beryllium reflector. They are numbered HB-1 through HB-12, and their layout is shown in Figs. 10 and 10A. Two different geometrical relationships with the core can be noted. Six beam holes are tangential to the core; the remainder are approximately radial. The tangential holes give experimenters a beam containing a smaller fraction of uncollided, high-energy neutrons and core gammas, but with some loss in overall flux in comparison with the radial holes. In elevation, four holes are 7 in. above the core midplane and the rest on the midplane. Table 7 lists the flux at the horizontal beam holes and other experimental ports.

Each beam hole has a minimum of 5 ft of perimeter at the surface of the reactor shield. The distance from shield to building wall varies from 32 to nearly 50 ft because the reactor is off-center to give some beam experiments extra clearance. The shape of the beam hole penetration within the reactor shield progressively changes to provide steps for radiation shielding (see Fig. 11). Starting at the outer surface of the shield, the dimensional changes are as follows:

<u>Distance</u>	<u>Cross Section</u>
Surface to 4 in.	66 in. x 48 in.
4 in. to 41 in.	18 in. x 18 in.
41 in. to 66 in.	17 in. x 17 in.
66 in. to 90 in.	24-in. diameter shutter
90 in. to 96 in.	13.5-in. diameter

The large indentation for the first 4 in. of the beam port is made so that external shielding for experiments can be recessed to minimize radiation leakage at the reactor face. Up to the shutter, the beam port is filled with removable shielding that will normally have penetrations specific for a given experiment. The shutter, which is 24 in. in diameter, is suspended on bearings from the top of the reactor shield through a full-diameter concrete-filled plug. The beam portion of the shutter section, which is removable, is of stainless-steel-boral construction, water cooled, and it has a 4 to 6-in. (depending on beam size) ID horizontal hole penetrating it.

FIG. 10
HORIZONTAL BEAM FACILITY LAYOUT

FIG. 10-A
REACTOR PLAN
SHOWING HORIZONTAL AND VERTICAL FACILITIES

This stopcock type of shutter arrangement allows transmission of the neutron beam only when the axis of the hole coincides with that of the beam. Rotation of the shutter by 90 degrees will cut off the beam. The rotational motion as well as the raising and lowering of the shutter is effected by mechanisms located within the top portion of the reactor shield (see Fig. 12). A 3-in.-ID hole through the center of the shutter, extending from the top of the reactor, is an added feature for beam monitors, filters, or samples. This hole extends through the lower shield to the basement.

Table 7

MAXIMUM UNPERTURBED NEUTRON FLUX IN EXPERIMENTAL FACILITIES

Number	Position	ID (in.)	Neutron Flux [neutrons/(cm ²)(sec)]	
			Thermal	Fast (above 0.625 ev)
VT-1	Vertical thermal column	4	5×10^{15}	
VT-2	Vertical thimble Be reflector	1	5×10^{14}	6×10^{13}
VT-3		1	5×10^{14}	6.3×10^{13}
VT-4		1	3×10^{14}	2×10^{13}
VT-5		1	3×10^{14}	2×10^{13}
VT-6		1.5	8×10^{14}	1×10^{13}
VT-7		1.5	8.5×10^{14}	9×10^{13}
VT-8		1.5	8×10^{14}	1×10^{14}
VT-9	Hydraulic rabbit	1.0	1.1×10^{15}	2×10^{14}
VT-10	Hydraulic rabbit	1.0	1.1×10^{15}	2×10^{14}
VT-11	Vertical thimble H ₂ O reflector	3	$1-7 \times 10^{13}$	$0.7-6 \times 10^{12}$
Through VT-21		3	$1-7 \times 10^{13}$	$0.7-6 \times 10^{12}$
HB-1	Horizontal beam tube	1.5 x 2.75	1.1×10^{15}	2.0×10^{14}
HB-2		1.5 x 2.75	1.8×10^{15}	3.8×10^{14}
HB-3		1.5 x 2.75	1.2×10^{15}	3.0×10^{14}
HB-4*		1.5 x 2.75	8×10^{14}	1.2×10^{14}
HB-5*		2 x 4.5	9×10^{14}	1.5×10^{14}
HB-6		2 x 4.5	1.5×10^{15}	4×10^{14}
HB-7		2.25 x 6.25	1.5×10^{15}	4×10^{14}
HB-8		4.25 dia	1.1×10^{15}	2.0×10^{14}
HB-9		4.25 dia	1.8×10^{15}	3.8×10^{14}
HB-10		1.5 x 2.75	1.2×10^{15}	3.0×10^{14}
HB-11*		1.5 x 2.75	8×10^{14}	1.2×10^{14}
HB-12*		1.5 x 2.75	9×10^{14}	1.5×10^{14}
HT-1	Horizontal through hole	4.25 dia	3.5×10^{14}	3×10^{13}
HT-2	Horizontal through hole	4.25 dia	3.5×10^{14}	3×10^{13}

*Holes above core midplane

FIG. 11
HORIZONTAL BEAM & THRU TUBE DETAIL

FIG. 12
REACTOR ELEVATION SHOWING BEAM HOLE SERVICE FACILITIES

Service facilities (such as collimator cooling) reach the inner beam hole components through a 3-in.-diameter duct from the basement that intersects the beam hole at a point 26 in. from the reactor face (see Fig. 12).

At the inner surface of the concrete shield the beam port narrows to a cylinder 13.5 in. in diameter. A bellows seal bridges the water gap between reactor vessel and shield, and a beam tube assembly, of 7-in. OD, continues to the beryllium zone inside the reactor vessel.

Some of the design features of the horizontal beam ports are summarized as follows:

1. Shutters - A choice of shutter motion is available to the experimenter. The initial installation will operate the shutters with rotational motion, using an electric drill of fractional horsepower as an operator. Reversing is not necessary. Any special-purpose actuator may be used interchangeably.

The shutter end plug may be changed to suit a particular experiment for which special materials or arrangements may be beneficial. Only the end portion need be replaced.

The initial costs of cylindrical sections are less expensive to fabricate and align.

Vertical access to the neutron beam in the shutter for monitors and services is possible.

2. Maintenance - All fittings which might leak water into the beam are replaceable without drainage of the reactor or the shield water systems. This includes replacement of a beam tube or a bellows (vessel to shield) seal.

3. Safety - A system to check beam tube leaks is incorporated into the design. Secondary beam tube seals are provided to back up tube damage.

B. Beam Tubes

Within the reactor a variety of sizes are used for the innermost portion of the beam tubes. Of two shapes, elliptical and cylindrical, two of the inner tube ends are 4.25-in.-OD cylinders with hemispherical ends and ten are elliptical sections with ellipsoidal ends. Seven of these measure 1.5 in. x 2.75 in.; two 2.0 in. x 4.25 in.; one 2.25 in. x 6.25 in. in diameter.

The choice of the elliptical section permits maximum utility of the beam shape in accordance with minimum loss of the beryllium moderator (an important item with a small core). Also, the ellipsoidal and spherical

end section design is stronger than a disc; hence, a reduction in thickness over the "window" section of the beam is possible.

In an effort to obtain the maximum flux for the experiments, all beam tubes terminate in or as near as possible to the flux peak in the beryllium reflector. To minimize parasitic capture, aluminum, beryllium, and zirconium were considered for the beam tube structural material. Aluminum end tubes are favored for conventional design, but there is some hope that beryllium may be a feasible alternate.

The wall thickness of the beam tube is reduced by internal pressures to compensate for the external static pressure of the reactor coolant. Although a second seal or window is required to pressurize the beam tube, it is in a zone where radiation damage is not a problem, so that a net savings in the combined mass of the "windows" is realized. Some additional benefits of beam tube pressurization are: a) an effective double seal to loss of reactor coolant in case of beam tube rupture; b) a continuous check on the integrity of the beam tube and its reactor seals at all times; and c) the pressurized helium gas will conduct heat from possible hot spots to give a more uniform temperature distribution in the beam tube. The beam tube ends are so designed as to give a 15 percent margin of safety to the yield strength of the weakest structural member in the event of loss of internal pressure. Loss of beam tube pressure would initiate immediate operational investigation, including a power setback.

A beam tube consists of an inner head section bolted and gasketed to a main body section fitted with a window seal plate (see Fig. 10). The main body is made of stainless steel and has an integral ring flange located near its axial midsection. The bolted flange is both a support and seal to the reactor vessel. Metallic gasketing is used throughout. The head section is bolted to the main body and is removable for replacement, dimensional changes, or repair. The experimenter will always have some degree of flexibility in the design of replacement tube headsections. The inner volume of the stainless steel main body section will accommodate a collimator with its coolant system. The coolant lines emerge through the outermost section of beam tube assembly adjacent to the window seal plate along with the beam tube pressurizing lines.

C. Experimental Service Facilities

The experimental service facilities are listed in Table 8.

Table 8

EXPERIMENTAL SERVICE FACILITIES

<u>Shutters</u>	Shutters for all ports
Type	cylindrical
Motion	Vertical rotation and/or raise and lower
Features	3-in.-ID monitor port in shutter 3-in.-ID access to basement
<u>Services</u>	
electrical (independent)	110 v, 30-amp, single phase
(available)	220 v, 15-amp, 3-phase
(available)	440 v, 3-phase
water	distilled
radioactive	recirculating
	cooled
nonactive	recirculated
	cooled
lab water	recirculated
	cooled
gas	
natural	as needed
air	lab air (60 psi)
He	Collimator He supply (250 psi)
vacuum	none
drainage: radioactive	supplied
nonactive	none
ventilation	radiological filter stack manifold vacuum manifold 3 in. Hg
<u>Beam tube equipment (see Fig. 13)</u>	
Size: OD	26 in.
Length	84 in.
hole size	10 in. ID
Weight	16,000 lb

D. Vertical Sample Irradiation Holes

A total of 21 sample positions are defined, but additional experimental volume will be available at lower flux levels in the water shield region outside the reactor pressure vessel. Figure 14 shows construction details of a typical vertical thimble. The design requirements are listed in Table 9.

FIG. 13
HORIZONTAL BEAM SERVICE EQUIPMENT

FIG. 14
VERTICAL THIMBLE DETAIL

Table 9

MAJOR DESIGN REQUIREMENTS FOR VERTICAL FACILITIES

		<u>No. of Vertical Facilities</u>
<u>Core</u> (Internal thermal column 1)		1
Size, ID	4 in.	
Coolant	Reactor system water	
<u>Be reflector</u>		7
Coolant	Regulated booster H ₂ O system or reactor water	
Size, ID	4 - 1 in. 3 - 1.5 in.	
<u>Hydraulic rabbits</u> (see Fig. 15)		2
Size, ID	1.0 in.	
Coolant	Regulated booster system	
Features	Position indicator	
<u>H₂O reflector</u>		11
Size, ID	3 in.	
Coolant	Booster or independent system	
Features	Access for instrumented samples

FIG. 15
HYDRAULIC RABBIT DETAIL

V. BUILDING LAYOUT AND CONSTRUCTION

A. General

Discussions with experimenters concerning their needs inside the reactor for beam and irradiation ports also developed an imposing list of requirements for the territory outside the reactor as well. Consequently, a number of layout studies were made to match these requests with the target budget and without compromising the experimental utility of the reactor. At the same time, some attention was given to the details of building construction in order to estimate the cost of various permutations in the plant.

In Table 10, the requirements for floor space, handling equipment, and overall plant construction are summarized. Floor plans are shown for the reactor building and laboratory in Figures 3, 16, and 17.

Table 10

BUILDING LAYOUT REQUIREMENTS

1	Floor Space Requirements	sq ft
a	Reactor and shield	
	(1) Reactor containment building basement	705
	(2) Reactor containment building main floor	705
b	Canal	
	(1) Reactor containment building main floor	540
	(2) Heat exchange building main floor	990
c	Primary cooling system including shielding, pipe trenches under canal, and space in heat exchange building	2300
d	Secondary cooling system in basement of heat exchange building	600
e	Control room on balcony in reactor containment building	600
f	Relay rack room on balcony below control room	600
g	Fuel storage vault in heat exchange building	400
h	Experimental work area	
	(1) Reactor containment building main floor	7500
	(2) Reactor containment building basement	3000
i	Active storage area in reactor containment building	2000
j	Storage in heat exchange building	2000
k	Equipment balcony in heat exchange building	900
l	Equipment area in heat exchange building including water treatment and misc. equipment	3250
m	Shop equipment in heat exchange building	600
n	Offices	
	(1) In laboratory building (9)	1476
	(2) In heat exchange building (2)	300
o	Laboratories	
	(1) In laboratory building (10)	3260
	(2) In reactor containment building (2)	500
p	Toilet	
	(1) In laboratory building (w/shower, locker and janitor's closet)	1295
	(2) In reactor containment building (w/shower)	150
	(3) In heat exchange building	150
q	Stock, shipping and receiving in laboratory building	512
r	Conference room in laboratory building	480
s	Drafting room in laboratory building	480
t	Counting rooms (2) in laboratory building	720

Table 10 (Cont'd)

u	Air conditioning equipment in laboratory building basement	2720
v	Observation balcony in reactor containment building (elevation operating floor + 24 ft)	1835
w	Experimental balcony in reactor containment building (elevation operating floor + 12 ft)	3000
x	Miscellaneous (corridor stairwells etc.)	
	(1) Laboratory building	4238
	(2) Reactor containment building	6960
	(3) Heat exchange building	4710
2	Floor Loading Requirement	Psf
a	Reactor containment building	
	(1) Basement	1500
	(2) Operating floor	1500
	(3) Control room and relay rack room	200
b	Heat Exchange building	
	(1) Basement	1500
	(2) Main floor	1000
	(3) Balcony	200
c	Laboratory building	
	(1) Basement	200
	(2) First and second floor	100
d	Stack building	500
e	Water treatment building	300
f	Canal floor	2000
3	Handling Equipment	
a	Reactor containment bridge crane (circular)	
	(1) Length	90 ft
	(2) Capacity (main hook)	40 ton
	(3) Capacity (auxiliary hook)	5 ton
	(4) Bridge speed	40 fpm
	(5) Trolley speed	40 fpm
	(6) Hoist (main) speed	2 speed 12/3 fpm
	(7) Hoist (auxiliary) speed	18 fpm
b	Heat exchange building bridge crane	
	(1) Length	60 ft
	(2) Capacity (main hook)	35 ton
	(3) Capacity (auxiliary hook)	5 ton
	(4) Bridge speed	40 fpm
	(5) Trolley speed	40 fpm
	(6) Hoist (main) speed	2 speed 12/3 fpm
	(7) Hoist (auxiliary) speed	18 fpm
c	Canal transfer car	
	(1) Capacity	5 ton
	(2) Speed	20 fpm
d	Reactor containment building equipment elevator	
	(1) Size	12 ft x 12 ft
	(2) Capacity	20 ton
	(3) Type	hydraulic platform
	(4) Height of travel	16 ft
	(5) Speed	Up 30 fpm Down 40 fpm
e	Laboratory building elevator	
	(1) Size	8 ft x 8 ft
	(2) Capacity	10 ton
	(3) Type	hydraulic platform
	(4) Height of travel	24 ft
	(5) Speed	Up 25 fpm Down 60 fpm
4	Special Equipment	
a	Air lock doors in reactor containment building	
	(1) Cargo door airlock	10 ft x 15 ft long x 12 ft high
	(2) Personnel door airlocks	4 ft x 10 ft long x 7 ft high
	(3) Pressure equalization system on reactor containment building	
	(a) Maximum interior pressure differential	(+) 1.5 psi
	(b) Maximum interior pressure differential	(-) 0.5 psi
b	Canal	
	(1) Length and width in reactor containment building	6 ft x 45 ft
	(2) Length and width in heat exchange building	8 ft x 55 ft
	(3) Depth	20 ft
	(4) Lining	stainless steel
c	Reactor Personnel Elevator	
	(1) Size	4 ft x 6 ft
	(2) Capacity	5 ton
	(3) Speed	Up 40 fpm Down 60 fpm
	(4) Travel	Basement to control room

FIG. 16
BASEMENT FLOOR PLAN

FIG. 17
2ND FLOOR OF LABORATORY BUILDING
& PLAN OF FLOOR OVER HEAT EXCHANGE EQUIPMENT

B. Plant Construction

1. Major Buildings

The characteristics of the major buildings are given in Table 11.

Table 11

<u>BUILDING CHARACTERISTICS</u>					
	<u>Reactor Containment</u>	<u>Heat Exchange</u>	<u>Laboratory</u>	<u>Stack</u>	<u>Water Treatment</u>
a Basement area, gross (ft ²)	11,300	10,200	5259	-	-
b First floor area, gross (ft ²)	11,300	4,560	4961	3,500	900
c Second floor area, gross (ft ²)	Balcony and control 4,200	660	4961	-	-
d. Basement ceiling height (ft)	15	60 and 24	12	-	-
e First floor ceiling height (ft)	75 to top of dome	38	12	16	10
f Second floor ceiling height (ft)	Control room 10	-	12	-	-
g Total volume, gross (ft ³)	900,000	525,600	185,000	56,000	9,900
h Foundation construction	Concrete	Concrete	Concrete	Concrete	Concrete
i Exterior wall construction	Steel dome	Steel	Brick	Steel	Steel
j Interior wall construction	Metal	Metal	Brick	-	-
k Roof construction	Steel dome	Metal pan	Concrete	Metal pan	Metal pan
l Floors construction	Concrete	Concrete	Concrete	Concrete	Concrete
m Stack height (ft)	-	-	-	175	-
n Stack material	-	-	-	Steel	-
o Air conditioning	yes	no	yes	no	no
p Windows	no	no	yes	no	yes
q Toilet and shower	MEN	MEN	MEN and WOMEN	no	no
r Cranes	40 Ton 5 Ton	35 Ton 5 Ton	-	-	-

2. Service Structures and Facilities

The service structures and facilities are given in Table 12.

Table 12

SERVICE STRUCTURES AND FACILITIES

Water tower	20,000 gal; 120 ft high (above core C)
Cooling tower (4)	50 ft x 50 ft x 47 ft high
Retention basins (4)	35 ft x 80 ft x 6 ft deep
Substation	30 ft x 30 ft
Parking lot	50 cars
Fencing	enclose entire site

3. Utilities

All utilities provided by the Laboratory's utilities system will be extended to service both building and process requirements.

a. Domestic water will be supplied for drinking fountains, toilet and shower rooms, janitor closets, and fire standpipe systems.

b. Laboratory water will be supplied to all process areas and equipment.

c. The ANL cooling-water system will be extended to provide makeup water for the cooling tower. Water will be chemically treated at the AARR cooling water-treatment building.

d. A sanitary sewer will service drinking fountains, toilet and shower rooms, and janitor closets. Where activity is involved, lines will be run to the retention tanks.

e. The Laboratory waste sewer will service all drains in the process areas, hot showers, and special equipment where activity is present. In the majority of instances, these drain lines will run to the retention tanks and, hence, to the Laboratory waste sewer system.

f. The storm sewer system will provide adequate storm water drainage and will be extended by gravity to natural drainage.

g. Natural gas will be supplied to the required areas.

h. Steam will be connected to the existing system and run to this facility for heating and processing.

i. The site electrical service will be extended to a substation at the reactor site.

j. Roads and sidewalks will be provided as required.

C. Shield Design Radiation Levels

The reactor plant shielding is discussed in detail in Chapter IX, where the computational methods used are given. The allowable dose rates in the various plant areas that were used to set the shield requirements are given in Table 13 for normal operation, shutdown, and emergency situations.

Table 13
ALLOWABLE DOSE RATES

Reactor at Power

1	Inner face of pressure vessel	
a	Total dose	5×10^{13} Mev/(cm ²)(sec)
b	Fast flux above 1 Mev	8×10^9 n/(cm ²)(sec)
2	Inner face of magnetite concrete biological shield	2×10^{11} Mev/(cm ²)(sec)
3	Outer face of biological shield at shortest diameter on reactor midplane	0.25 mr/hr
4	At surface of canal water above the reactor	0.25 mr/hr
5	Outer surface of primary coolant pipe tunnel at the base of the spent fuel transport canal	0.25 mr/hr
6	Spent fuel storage and handling canal	
a	Outer surface of wall opposite primary coolant heat exchanger cubicle	4 mr/hr*
b	Water surface	
	From spent cores	2 mr/hr*
7	Heat exchanger building	
a	Floor above all heat exchanger and coolant cleanup system cubicles	2.5 mr/hr
b	Surface inside of cubicle wall for radiation originating in adjacent cubicle	7.5 mr/hr
c	Walkway side of wall separating walkway from coolant cleanup systems	7.5 mr/hr
d	Exterior surface of primary coolant heat exchanger cubicle building wall	7.5 mr/hr
e	Exterior surface of primary coolant cleanup system cubicle building wall	2.5 mr/hr

Reactor during Shutdown Period under Normal Conditions

1	Canal water surface above reactor from 2-hr decay core	15 mr/30 sec
2	Control rod drive room directly below steel shield 15 min after shutdown	10 r/hr

Immediately after Fuel Meltdown Accident at All Personnel Locations

Shielding Casks

1	One meter from any readily accessible cask surface	
a	Californium ²⁵² transfer cask	10 mrem/hr
b	Spent-fuel shipping cask	10 mr/hr
2	Cask exterior surface	
a	Californium ²⁵² transfer cask	200 mrem/hr
b	Spent-fuel shipping cask	200 mr/hr

*These dose rates apply during shutdown after a 2-hr old core is placed in the canal

D. Plant Containment

1. Design Criteria

There are three major sources of energy that can internally load the AARR containment structure. They are the stored thermodynamic energy in the primary coolant, the energy from a nuclear excursion plus fission product decay, and the chemical energy from a metal-water reaction. Some of the energy storage and release values pertinent to an evaluation of containment requirements are as follows:

Energy from a nuclear excursion large enough to melt core	87 Mw-sec
Energy from a complete core metal-water reaction	184 Mw-sec
Energy stored in primary cooling water at 190°F with air at 80°F	29000 Mw-sec
Energy needed to raise water in the reactor vessel to 212°F	392 Mw-sec
Energy needed to raise canal water 1°F	792 Mw-sec
Energy needed to raise primary water 1°F	264 Mw-sec

It is apparent that the largest source of energy release in a serious accident can be from heat stored in the primary coolant system. In the AARR design, this potential source of energy release is minimized by selection of operating conditions that will keep the bulk water temperature at the exit below 212°F, so that flashing will not occur. Also, the canal water can be used as a heat sink to reduce internal loading further in the event of an accident. The resultant internal pressure load on the containment structure from release of the primary water and using the canal water as a heat sink is 1.31 psig.

2. Design Pressure and Shell Construction

Negative pressure due to changes of barometric pressure of 0.50 psi and to snow and wind loads govern the design. A relief valve will be wide open at $\frac{1}{2}$ psi. The design load is 0.75 psi.

The containment building consists of a cylinder with a hemispherical dome and a flat bottom. The dome and part of the cylinder is of steel. The remainder of cylinder and the floor is concrete. There are two steel stiffener rings on the steel part of the cylinder. The data are summarized in Table 14.

Table 14

DESIGN PRESSURES AND SHELL CONSTRUCTION

Internal design pressure	1.5 psi
Steel plate thickness	$\frac{1}{2}$ in.
Concrete wall thickness	2 ft
Concrete floor thickness	8 in.
Stiffening rings on steel cylinder (Two required, 8-ft. separation)	21-in.-wide flange (59#)
Permissible leak rate	$\frac{1}{4}$ % of total building volume per day at 1.5 psi
Applicable Code	ASME Boiler and Pressure Vessel Code for Unfired Vessels, Section VIII
Additional containment provisions	The door in canal will be closed during the operation of reactor to complete containment shell
Height of dome above basement floor	105 ft
Diameter of hemispherical dome and cylindrical shell	120 ft
Height of cylindrical part of shell	16 ft
Height of concrete	29 ft
Total cylindrical height	45 ft
Materials of construction	
Steel plate	A-201 Grade B Firebox with Charpy values as prescribed in ASME Boiler and Pressure Vessel Code and the applicable nuclear code cases
Concrete	Free of cracks or sealed.

3. Fuel Loading

The current concept calls for moving the entire core as a unit, with the option of changing or adding single fuel boxes as desired. Preparation of a new loading starts with the dry, bench assembly of the fuel elements, control blades, and bottom support structure. This unit is then checked for critical and control properties by zero-power tests conducted in a special section of the canal. Afterwards, the core assembly is moved to the reactor vessel and the control drive rods connected to the poison blades which remain locked in the core during assembly and transportation.

To unload a core, the reverse of the loading procedure is used. However, the situation is complicated because the core is now "hot," both radioactively and thermally. The following two situations will be considered: first, a normal core removal; second, a core removal after a cladding failure.

For normal core removal, the procedure to be used is as follows. The canal dam is placed in position to prevent contamination of the entire canal should the core prove to have been damaged and fission products released. After the primary system pressure is reduced to atmospheric, the pressure vessel top is removed. The irradiation sample is removed from the central flux trap and stored in a coffin in the canal. Additional poison strips are then inserted in the core, and the control rod drives are disengaged. The core is lifted from the pressure vessel with a removal fixture and placed in a core-storage container which rests on the canal service car.

The core-storage container will be designed to prevent the spread of any contaminants from the core and at the same time provide for an adequate supply of water to keep the core cool. The container will be a cylindrical can having a removable top cover. Boron stainless steel is specified for construction to reduce the criticality hazard. The container will be designed so that if any scale or foreign material (which could be highly radioactive) is dislodged from the fuel, it will drop to the bottom of the container. The inlets for the cooling water will be above the bottom of the container and will allow the foreign material to lie undisturbed on the bottom. The cover will have an opening in it to allow the water which has been heated by the decaying core to mix with the canal water.

Once the core has been placed in the storage container, the canal dam can be removed and the canal service car can transport the container to the heat exchange building for storage until such time as the core has decayed enough to permit dismantling.

Two means are considered to insure an adequate cooling while the core is in the container. One is to use a chimney or stack to confine the heated water and release it to the canal water several feet above the core and thus to increase the differential head for natural circulation through the core. The other method is to rig a line from the top of the container to the suction side of the pump that is used for the canal cooling and cleanup system. This will provide a positive flow through the core and purify the water before it is recirculated to the canal. This second method is used when it is necessary to prevent contamination of the canal by a damaged or dirty core.

For the problem of core removal when a fuel plate has ruptured, the procedure to be used is similar to the foregoing method. It must be recognized that if a gross fuel plate failure occurs, contamination will rapidly spread throughout the primary system. Since a period of several hours is required to cool the core before the primary system pressure can be reduced and steps taken to remove the core, a major cleanup of the primary system will be required.

Assume that the monitoring system for primary water system has indicated a rise in activity and that the proper steps have been taken to secure the reactor (by either automatic or manual means). A period of several hours must elapse until the decay heat in the core has decreased sufficiently so that adequate cooling of the core during the removal process is maintained by natural convection. As soon as the pressure in the primary system is relieved and the pressure vessel top removed, the process of core removal can start. Prior to removing the pressure vessel closure, however, the canal dam is positioned to isolate the reactor section of the canal. Upon removal of the pressure vessel top, the primary water will mix with the water shield above the reactor. If contamination is serious, it may be necessary to extend the normal time between reactor shutdown and opening of the pressure vessel in order to reduce the radioactivity in the primary water.

Next, the entire core is removed and placed in a core-storage container as outlined in the procedure for a normal core removal. For this case, however, the container will be connected to the canal cooling and cleanup system to limit the spread of contamination. The core would be left in this container until such time as it was to be disassembled.

After the core has been removed, the pressure vessel top is replaced and the primary water is circulated through the primary water cleanup system. Water from the section of the canal above the reactor which was contaminated by the primary water is bypassed through the canal cleanup system. By the time the core has been placed in its container and the pressure vessel closure replaced, the activity of the shield water should have been reduced sufficiently to permit removing the canal dam and transferring the core-storage container to the canal section in the heat exchange building.

The design of the AARR also provides for the possibility of a major fission break or partial meltdown of a fuel element. The distribution of contamination can be reduced by valving off various components in the primary system. Each major component in the primary system has a shutoff or isolation valve. Thus, in the preceding type of emergency, all but one of the heat exchangers and one pump may be isolated from the primary system in an attempt to limit dispersal of radioactivity. The isolation valves cannot be closed fast enough to eliminate contamination, but the level is reduced significantly.

A trap is provided in the primary system at the pressure vessel exit to stop particles from entering the heat exchangers and the pumps; it is also possible for pieces of fuel to fall to the bottom head of the pressure vessel. These must be retrieved after the core has been removed.

After reactor shutdown and subsequent isolation of primary system components, the water cleanup system is put into operation. Additional shielded ion exchange columns will be available for emergency use to help decontaminate the water in the primary system. If it is necessary to flush or purge with deionized water, the effluent will be stored in the retention system until it can be processed and dumped. Some purging of the primary system may prove desirable even under normal conditions.

VI. PLANT VENTILATION, HEATING, AND AIR CONDITIONING

The design criteria for heating, ventilating and air conditioning of the reactor complex are based on established safety practices for the protection of Laboratory personnel in areas where radioactive materials are encountered.

For example, air flow is from clean areas inward toward potentially contaminated areas to prevent the spread of radioactivity. Where adjoining areas both contain radioactive materials, air flow is toward the area containing materials producing the highest potential radiation hazard. This flow control is applied not only to the direct mechanically induced ventilation, but also to the infiltration of air from one area to another.

An outline of the functional requirements for heating, ventilating, and air conditioning follows:

A. Heating

1. Reactor Building

The entire reactor building will be heated by the ventilation air. Reheat coils will be provided for the various zones where direct supply air is introduced. In areas where no direct supply air is introduced, low-pressure steam coils will be provided in fan-coil circulating units located within the space. The space temperature will be maintained at $75^{\circ}\text{F} \pm 2^{\circ}\text{F}$ with an outside design temperature of -10°F . The main supply air intake and preheat coils will be located in the service floor of the laboratory building. From here, steam lines will pass through the containment shell to supply the reactor building. Automatic isolation valves will be provided that will close in the event of a major reactor incident. Return condensate will be pumped to the service floor of the laboratory building.

2. Stack Building

The stack building will be heated by steam unit heaters supplied by steam and condensate return overhead lines from the site main.

3. Equipment and Heat Exchange Building

Air heating will be provided by separate reheat coil for each level of the building. Each coil also will be thermostatically controlled to meet the demands of the respective level served. The building is to be maintained at 75°F when it is -10°F outside with a 15-mph wind prevailing.

4. Laboratory Building

Heat for the laboratory building will be supplied by console units in each room. These same units will also serve for comfort cooling. As required, hot or chilled cold water will be circulated through the coil section. Each unit will be individually thermostatically controlled and will be provided with manual dampers. The space temperature will be maintained at 75°F with an outside design temperature of -10°F. Some areas may be heated directly with fin-tube radiation units as required.

B. Ventilating

Proper ventilation for the major buildings of the facility requires a correctly controlled, filtered air supply system. Air supply for the reactor building and for the equipment and heat exchange building will be filtered through primary and secondary glass media filters of the low-pressure drop type.

1. Reactor Building

The air supply equipment for the reactor building is on the service floor of the laboratory building, (inlet air filters, preheat coils, cooling coils, reheat coils, and fans). Here, also, will be located the isolation valves at the entrance of the air supply ducts to the reactor building. These valves will close in the event of a reactor incident.

The air supplied to the reactor building will flow from areas of least potential hazard to areas of successively higher potential hazard.

2. Equipment and Heat Exchange Building

The air-handling unit will consist of filters, preheat coils, and fans. Distribution will be outward from areas of least potential hazard.

3. Stack Building

Roof exhausters will remove heat from internal equipment. Air-intake louvers will be installed in walls at suitable locations. Both fans and intake louvers will be operated by thermostatic control.

4. Laboratory Building

The office area will be ventilated by means of fan-coil console units in each room. Direct outside air will be mixed with room air and circulated through the units. The service floor area will be provided with summer ventilation to cool operational equipment. Locker and work room facilities will be ventilated according to standard practice.

C. Air Conditioning

1. Reactor Building

All areas of the reactor building will be held to 80°F and 50% maximum relative humidity. Air will be cooled with chilled water coils located on the service floor area of the laboratory building. Humidity will be regulated by a dew point controller. Reheat coils will be provided for each of the three zones of the supply system to establish design temperature and humidity requirements. Separate cold water coils will be installed in the reactor basement because it is not cooled by the main supply air system.

Refrigeration for water cooling is to be provided by a centrifugal-type refrigeration machine with a multicell, winterized cooling tower. It will serve both the reactor building and the laboratory building. The cooling tower, located west of the reactor building, will be designed for outdoor summer conditions of 95°F dry bulb and 80°F wet bulb.

2. Stack Building

No air conditioning will be provided.

3. Equipment and Heat Exchange Building

No air conditioning will be provided.

4. Laboratory Building

Console units in each room will maintain an inside temperature of 80°F within a 45-55% range of relative humidity. The central chilled water refrigeration unit will be located on the service floor with a winterized water-cooling tower to service the unit located outdoors at grade. Refrigeration load for this building is 50 tons. No air conditioning will be provided for the service floor area.

D. Exhaust Systems

The exhaust systems for the major buildings will be designed for the safe removal and containment of radioactive materials. Special emphasis will be placed on control of air volume and flow patterns. Established practices dictate the use of prefilters and AEC final filters for all exhaust air from areas of potentially high radioactive contamination.

1. Reactor Building

Three exhaust systems will serve the reactor building. A general air exhaust from the main floor and basement will be delivered to plenums ahead of final filters in the stack building. The prefiltered exhaust

of the active storage will be delivered to this same plenum by means of booster fans.

A beam hole exhaust system will be prefiltered through high-efficiency filters and delivered to the main plenum by means of exhaust booster fans, one of which will be a standby fan. The beam hole exhaust system will handle a varying amount of air depending upon additional experimental loads and other local requirements about the surface of the reactor. All prefilters will be located in the basement. Each of the exhaust ducts delivering air to the plenum located in the stack building will be provided with an isolation containment valve at point of exit from the reactor building.

2. Equipment and Heat Exchange Building

Three exhaust systems will be provided: two will serve the operating areas; the third will remove heat from the primary and secondary coolant pump motors. The general air exhaust will take air from the main floor and deliver it to the plenum of the filter bank in the stack building. The air indirectly delivered to the primary water equipment area will serve for ventilation of experimental facilities, emergency pump room, coolant pump rooms, and heat exchange rooms. Air exhausted from these rooms will be prefiltered and delivered to the plenum of the main filter bank through the exhaust boosters for prefiltered exhaust system. The air exhaust from the equipment building will enter common ducts from the reactor building that run above ground to the stack building.

3. Stack Building

This building will house the filter bank for final filtering of all exhaust air and the two booster fans for all prefiltered air from the reactor and the equipment building. (The exhaust booster fans for the special absolute prefilters for beam hole exhaust will be located with the prefilters in the basement of the reactor building.) The final filter bank will be provided with two stack exhaust fans, each to carry half the required total load. One will be on the emergency electrical system, but either will have a capacity in excess of the total prefiltered air exhaust. Both fans will be connected to the 5-ft-diameter, 175-ft-high stack through a manifold.

The final filter bank will consist of an inlet plenum, AEC filters, and an exit plenum. Each filter cell will be connected by ducts to the plenums at each end of the rows.

VII. PHYSICS

Since the results of the critical experiments have been reported in detail only a summary is given here. Most of this section will cover the physics calculations.

A. Critical Experiments

The ZPR-VII facility, previously used with the THUD program, was converted to run critical experiments for AARR. The core structure with beryllium reflector penetrated by one beam hole at 3 o'clock is shown in Fig. 18. Cruciform poison sections placed at the beryllium-fuel interface were used for control during the critical experiments, instead of blades which will be specified for the reactor design. Otherwise, the core as shown duplicates the core composition intended for the reactor.

Fig. 18

Critical Experiment Core Region

In addition to the usual critical measurements, flux and power distributions were found for cores of approximately cold clean initial (C-2) and end-of-life (C-1) compositions, respectively. Each core included a 100-liter annular fuel zone, 50.8 cm in height, enclosing a 12.7 x 12.7 x 50.8 cm internal thermal column filled with H_2O . The external radial

reflector consisted of beryllium blocks stacked on plexiglas spacers to yield approximately 90% beryllium, 10% H₂O and plexiglas. The top and bottom reflectors were effectively infinitely thick H₂O. The fuel contents of the cores were 7.25 kg (C-2) and 3.64 kg (C-1) U²³⁵ (highly enriched uranium) contained in 40 fuel clusters. Core C-2 included 34.34 kg Type 304 stainless steel evenly distributed to simulate burnable poison. The remaining metal was aluminum. The metal-to-water volume ratios were 1.60 (C-2) and 1.16 (C-1) in the fuel region.

The ratios of maximum-to-average radial power distributions in the midplane of the fuel region were 2.42 (C-2) and 2.05 (C-1). The ratios of maximum-to-average power along the axis of various clusters ranged from 1.25 to 1.46.

The measured effects of physical variations included, among others, the distributed void coefficient (displacement of H₂O), and fuel and poison worths, all of which depended upon location in the core. The temperature coefficient, cadmium ratios with several materials, prompt neutron life-times, control rod calibrations, and various other measurements were made. In the internal thermal column, the measured thermal neutron flux peak per watt was 3.1×10^7 n/(cm²)(sec) for the clean core and 2.8×10^7 n/(cm²)(sec) for the end-of-life core.

Table 15 lists the more important lattice parameters that were measured along with some calculated values for comparison.

Table 15

CHARACTERISTICS OF CRITICAL ASSEMBLIES

Item	Core	Result	How Obtained
Critical mass	C-1	3.58 kg	Experimental
	C-2	7.15 kg	Experimental
Prompt neutron lifetime	C-1	140 μ sec	Calculated
	C-2	71 μ sec	Calculated
	C-1	43 $\pm$ 3 μ sec	Pile Noise
	C-2	33 $\pm$ 2 μ sec	Pile Noise
	C-2	73 μ sec	Ross α
Safety rod worth	C-1	1.5% $\delta k/k$	Calculated
	C-1	1.56% $\delta k/k$	Measured
	C-2	2.0% $\delta k/k$	Measured
Fine rod worth	C-1	0.8% $\delta k/k$	Calculated
	C-1	0.7% $\delta k/k$	Measured
	C-2	1.0% $\delta k/k$	Measured
Peak 2200 m/s flux per watt	C-1	2.8×10^7 n/(cm ²)(sec)	Measured
	C-2	3.1×10^7 n/(cm ²)(sec)	Measured
Over-all void coefficient	C-1	-0.37% $\delta k/k/\%$ void	Measured
	C-2	-0.34% $\delta k/k/\%$ void	Measured
Temperature coefficient	C-1 and C-2	-0.24% $\delta k/k/^\circ\text{C}$	Measured

NOTE: C-2 is the cold clean core; C-1 is the end-of-life core.

B. Theory

1. Physical Configuration

Physics calculations were made for a variety of configurations and compositions; the greatest attention was given to the array shown in Fig. 19. The atomic densities assumed in the compositions of the various regions of Fig. 19 are given in Table 16. The heights of the regions shown in Fig. 19 were taken to be uniformly 45 cm. In addition, it was assumed that there was 30 cm of water at the top and bottom of the regions 1-5.

Table 16

ATOMIC DENSITIES (atoms $\times 10^{24}$ per cm^3)

Region number (see Fig. 19)	Isotope						
	H	O	U^{235}	Al	B^{10}	Cd	Be
1 Sample*	0.0361	0.01821	0.9606×10^{-5}	0.02709	0.1947×10^{-5}		
2 Water	0.06620	0.03310					
3 Core	0.03310	0.01655	0.0002142	0.03010	0.5137×10^{-5}		
4 Control Rod (or)	0.03310	0.01655				0.0232	
4 Follower**				0.06020			
5 Outer Reflector	0.00662	0.003310					0.11124

*The sample composition mocks up the nuclear properties of a sample containing 100 gm of curium-242 and 1.5 gm of curium-245. Values of $\sigma_a(\text{Cm}^{242}) = 50 \text{ b}$, $\sigma_f(\text{Cm}^{245}) = 2000 \text{ b}$ have been assumed at 0.025 ev.

**In some problems a mixture of aluminum and water was used to represent the follower.

2. Mode of Computations

a. Cross Sections

Four-group diffusion theory was used for most calculations; neutron transfer cross sections were used to provide for transfer of neutrons from a given group to the next lowest group only. The group structure and assumed fission spectrum are given in Table 17. The treatment of the thermal group is somewhat arbitrary. Code limitations as to the number of groups and as to the use of upscattering are such as to make a reasonably complete treatment of the thermal group an unreasonably expensive and tedious process when a large number of problems is involved. Since the PDQ codes⁴ were suitable for most of the physics problems considered, it was decided to use just one thermal group. The thermal group cross sections were found by using the Sofocate code⁵; in regions 1-4 the scattering matrix was that of a monoatomic gas of mass 1. In region 5 (the Be water-cooled reflector), the scattering matrix was computed on the basis of free hydrogen atom scattering and free beryllium atom scattering. No significant difference between the two methods was found. The latter method was the one finally adopted. A temperature of 400°K was assumed.

Table 17

GROUP STRUCTURE

Group	Upper Energy Limit	Lower Energy Limit	Lethargy Width	Fission Spectrum
1	10 Mev	1.353 Mev	2.0	0.575
2	1.353 Mev	9.12 kev	5.0	0.425
3	9.12 kev	0.625 ev	9.8	0
4	0.625 ev	0	-	0

⁴G. G. Bilodeau, W. R. Cadwell, J. P. Dorsey, and R. S. Varga, PDQ - An IBM 704 Code to Solve the Two Dimensional Few-Group Neutron-Diffusion Equations, WAPD-TM-70 (Aug 1957).

⁵H. Amster and R. Suarez, The Calculation of Thermal Constants Averaged Over a Wigner-Wilkins Flux Spectrum; Description of the Sofocate Code, WAPD-TM-39 (Jan 1957).

This treatment is in error at interfaces between regions. In general, the deviation of the actual spectrum from the infinite medium spectrum should not be large at distances larger than one relaxation length from the interface where the relaxation length in the medium is computed on the basis of the spectrum of the entering neutrons. This problem has been discussed by Kelber and Kier⁶ and by Michaels.⁷

A calculation of the power distribution in the core was made with only water in the internal thermal column. The maximum power in the core changed by 10% as the water thermal spectrum was changed from the infinite medium water spectrum to the core spectrum.

Fast-group cross sections were computed by means of the MUFT code.⁸ No resonance self-shielding was assumed. In general, no small-scale spatial disadvantage factors were used.

The four-group cross sections used are listed in Table 18. These cross sections are characteristic of the core at start of life.

Table 18

FOUR-GROUP CROSS SECTIONS

Material	Group	D(cm)	$\Sigma_{c+H}(\text{cm}^{-1})$	$\Sigma_{J+H}(\text{cm}^{-1})$	$\nu\Sigma_f(\text{cm}^{-1})$
Core	1	2 486689	0 001382	0 068011	0 000693
	2	1 318305	0 000467	0 062699	0 000766
	3	1 022132	0 009195	0 065775	0 012233
	4	0 4699	0 089710	-	0 1453
Water	1	2 265983	0 001849	0 106004	0
	2	1 176465	0 000009	0 123945	0
	3	0 608080	0 000899	0 149450	0
	4	0 19490	0 01580	-	0
Outer Reflector	1	2 104779	0 000169	0 099125	0
	2	0 651149	0 000001	0 040824	0
	3	0 493860	0 000090	0 028271	0
	4	0 35023	0 002306	-	0
Control Rod*	1	2 104779	0 000169	0 099125	0
	2	0 651149	0 000001	0 040824	0
	3	0 493860	0 028361	0	0
	4	0	0 469483	-	0
Sample	1	2 294775	0 001184	0 071819	0 000031
	2	1 282212	0 000090	0 066997	0 000034
	3	0 961530	0 001568	0 074390	0 000519
	4	0 359800	0 022020	-	0 008870
Follower	1	3 076165	0 000020	0 022880	0
	2	1 845699	0 000140	0 000120	0
	3	3 955070	0 000600	0 000480	0
	4	3 460686	0 00993	-	0

*This cross-section set is adopted for use with the PDQ code, the cross section of the control rod in group 4 is artificial

⁶C. N. Kelber and P. Kier; Few Group Analysis of D₂O-U²³⁵ Assemblies, Nuc. Sci. and Eng. 8 (1), 1-11 (July 1960)

⁷P. Michael, Thermal Neutron Flux Distributions in Space and Energy, Nuc. Sci. and Eng. 8 (6), 426-431 (Nov 1960)

⁸H. Bohl, Jr., E. M. Gelbard, and G. H. Ryan, MUFT-4, Fast Neutron Spectrum Code for the IBM-704, WAPD-TM-72 (July 1957)

b. Computational Methods

As mentioned previously, the PDQ codes⁴ were used for static computations. These codes solve the four-group diffusion equations in two dimensions. Both R-Z and X-Y geometries were used. A calculation was made to verify that, in going from X-Y to R-Z geometry, preserving the areas of the various regions preserved the reactivity, but that preserving the mean chord length in the internal thermal column preserved the peak thermal flux per unit power.

In addition, one simplified problem using only two groups and a coarse mesh was performed in three dimensions by means of the TKO code⁹ together with a corresponding problem in X-Y geometry to check the assumed value of the average of the axial maximum-to-average power generation over the core.

Burnup problems were studied by several means. Initially, hand computations of average isotopic burnup were coupled with PDQ calculations to yield estimates of the effects of various poisons. One-dimensional calculations using a radial geometry were performed with the Candle code.¹⁰ Finally, a two-dimensional problem was run using the TURBO code¹¹; in the latter, however, beryllium cross sections were not available in the cross-section library, so carbon was used with an atomic density of twice normal. This mockup was not successful.

In addition, an analog method^{12,13} was developed to evaluate time-dependent problems for a wide range of problems. In particular, this method was used to study the control problem following a fast reduction in power.

⁹W. R. Cadwell, TKO - A Three-dimensional Neutron-diffusion Code for the IBM-704, WAPD-TM-143 (Oct 1959).

¹⁰O. J. Marlowe and P. A. Ombrellaro, Candle - A One-dimensional Few-group Depletion Code for the IBM-704, WAPD-TM-53 (May 1957).

¹¹J. B. Callaghan et al., Turbo - A Two-dimensional Few-group Depletion Code for the IBM-704, WAPD-TM-95 (Nov 1957).

¹²L. C. Just, C. N. Kelber and N. F. Morehouse, Solution of Multiple Region Reactors Using a Minimum Number of Mesh Points, Trans. of Am. Nuc. Soc. 3(2), 340 (1960).

¹³C. N. Kelber, L. C. Just, and N. F. Morehouse, Jr., The Solution of Many Region Reactor Kinetics Problems on an Analog Computer, Submitted for publication in Nuc. Sci. and Eng.

3. Properties of the Clean Core

Table 16 lists the atomic densities in the various regions of the hot clean core. The initial core loading is 6.5 kg of U^{235} . In addition, B^{10} has been included to reduce the reactivity to approximately 1.0 with the control rods fully inserted. In these calculations, the central sample (region 1) is included, but the effects of the other experimental facilities are omitted. Thus the initial B^{10} loading is to be regarded as an adjustable parameter.

The reactivity worths of beam holes can be roughly estimated, but since the layout of the proposed experimental facilities was being made at the same time as the physics studies, an arbitrary allowance for experimental facilities of $6\% k_{eff}$ was made. In addition, $2\% k_{eff}$ was allowed for calculational error.

The four control rods shown as region 4 have a total worth of 10%, and there is little shadowing of one rod by another where they are symmetrically inserted; by itself one rod is worth 2.69%. In addition, safety rods inserted between the core and the beryllium reflector have a worth of 12% if the entire interface is covered. Table 19 lists the reactivity balance in the core.

Table 19

REACTIVITY BALANCE IN CLEAN, FRESH CORE

k_{eff} cold, clean:	1.125
cold to hot	-0.01
Experiments	-0.06
Error allowance	-0.02
Rods	-0.035
	<hr/>
	1.00

The thermal flux in the sample is 3.00×10^{15} neutrons/(cm²)(sec) at 100 Mw; the unperturbed flux at the sample position is 3.67×10^{15} neutrons/(cm²)(sec).

4. Long-term Burnup Properties

The Candle code was used to compute the properties of the core as it burnt up. Nonsaturating fission products were assigned an average absorption cross section at 2200m/s of 60b per fission product pair. The geometry used was radial, the effect of control rods being represented by the addition of uniform poison. Table 20 summarizes the results of the Candle problem. The power was assumed to be held constant at 100 Mw. The column headed k_{ex} gives the reactivity held down by the rods; 8% is assumed to be held down by experiment and error. The column headed

$P_{\max}/P_{\text{avg}}$ is the overall maximum-to-average power ratio; the axial maximum to average is assumed constant at 1.25 (see Section 5). The next column, headed ϕ_{th} , is the peak unperturbed flux at the sample position; the perturbed flux is lower than this by a factor of 1.22. The next-to-last column gives the peak unperturbed flux ϕ_{th}^* if the reactor were operated at constant maximum power rather than constant average power; this is a zero-order approximation since in such an operation the burnup between successive time steps would be about 10 to 15 per cent greater than in the case actually studied. Since the benefits of such operation depend on the overall duty cycle of the reactor, this mode of operation was not studied further.

The dip in k_{ex} as xenon comes in and the anomalous behavior of $P_{\max}/P_{\text{avg}}$ in the first two days of operation may be laid to the fact that B^{10} does not burn out fast enough to compensate for xenon; it may be desirable to substitute some finely divided cadmium for part of the B^{10} to compensate for the xenon.

Table 20

BURNUP CHARACTERISTICS
(operation at 100 Mw)

t(sec)	k_{ex}	$P_{\max}/P_{\text{avg}}$	$\phi_{\text{th}} (x 10^{-15})$ [n/(cm ²)(sec)]	$\phi_{\text{th}}^* (x 10^{-15})$ [n/(cm ²)(sec)]	U^{235} (kg)
0	0.035	2.6	3.67	3.82	6.500
10^5	0.0015				6.360
2×10^5	0.0033	2.7	3.70	3.70	6.221
3×10^5	0.0064				6.081
4×10^5	0.0087	2.5	3.72	4.02	5.942
5×10^5	0.0190				5.802
6×10^5	0.0103	2.35	3.73	4.29	5.663
7×10^5	0.0095				5.524
8×10^5	0.0078	2.18	3.75	4.64	5.384
9×10^5	0.0050				5.245
10^6	0.0011	2.01	3.76	5.05	5.106

NOTE ϕ_{th} is the peak unperturbed thermal flux (at the Cf sample position), the perturbed flux is lower by a factor of 1.22. ϕ_{th}^* is the approximate peak unperturbed flux which is obtained if the reactor is operated at constant maximum power rather than constant average power.

The Candle code (version I)[†] does not allow a change in thermal spectrum in time so that a small error is introduced this way; but since the total power is kept constant, the correct amount of U^{235} is fissioned per unit time. The total burnup will be slightly in error because of the change in

[†]Later versions of Candle permit an adjustment of element cross sections which eliminate part of this error; these versions were not in use at ANL when this work was done.

α , the capture-to-fission ratio in U^{235} , as will the reactivity. In both cases the error is a conservative one since α tends to get smaller with time. Candle predicts more burnup and lower reactivity with time than a time-dependent spectrum would.

The effects of geometrical configuration are harder to evaluate over the course of the cycle since they call for extensive computing machine time. The effects of geometry were, therefore, evaluated for the fresh clean core and are discussed in the next section.

5. Geometric Effects

The total vertical reflector savings was estimated to be 15 cm. This was verified by comparing the calculated reactivities of radially reflected cylinders of varying heights with the reactivity calculated in an R-Z geometry.

Calculations in R-Z geometry were then compared with calculations in the X-Y geometry (vertical savings of 15 cm were assumed) with respect to reactivity and to peak flux per unit power. The former, as expected, suffered very little change when the volumes of the various regions were preserved. The latter quantity suffered very little change when the mean chord length of the internal thermal column was preserved. This is reasonable: the fast neutron flux is very nearly constant over the internal thermal column; the total source for the thermal flux then is closely proportional to the average fast flux times the removal cross section times the mean chord length. The effects of thermal neutrons leaking in from the core are small since the thermal diffusion length in water is small.

In the Candle problem (Section 4) the mean chord length was preserved and the reactivity adjusted to be correct at the beginning of core life.

The effect on flux shape and reactivity of extending core aluminum into the upper and lower reflectors was studied and found to be negligible. In a more heavily loaded core, however, with more epithermal absorption, this conclusion does not necessarily follow.

The number of neutrons per unit total source, which become thermal at the center of the reactor and are born in fission in various parts of the reactor, was also studied. The method was to run a PDQ problem to completion and save the flux tape. Then an almost identical problem was run with the converged flux tape as input information and in which ν , the number of neutrons per fission, was set equal to zero except in the region of birth that was under consideration. The problem was run for just one outer iteration and with a tight internal convergence criterion,

and a flux edit made. The result was just the answer desired, since the source present has the same shape it has in the critical reactor.*

It was found that 70% of the neutrons which become thermal at the center arose from fissions in the innermost (radially) $\frac{1}{3}$ of the core, 20% of the neutrons arose from fissions in the middle $\frac{1}{3}$ of the core and 10% from the fissions in the outer $\frac{1}{3}$ of the core. This is of considerable consequence in discussion of power-flattening techniques.

Analysis of R-Z computations showed that the variation of the power in the Z direction departed considerably from a chopped cosine over the core. Therefore, a three-dimensional (X-Y-Z) problem was run and the results compared with an X-Y problem in which separability with respect to Z (and, therefore, a chopped-cosine distribution) was assumed. From this comparison an average over the core of the axial (Z direction) maximum-to-average power could be obtained, and was found to be 1.22. The chopped-cosine approximation yielded 1.25, and this was used in all other calculations.

6. Reactivity Coefficients and Lifetime

The reactivity coefficients have been estimated by computing the reactivity as a function of a given parameter with the others fixed. Thus the temperature coefficient does not include the effect of change of water density. The void coefficient is in terms of per cent void in the water. The lifetime was computed by the well-known $1/v$ method. The coefficients are summarized in Table 21.

Table 21

REACTIVITY COEFFICIENTS

Temperature	-2.3×10^{-5}	$\delta k/^{\circ}\text{K}$ in range $380^{\circ} < T < 400^{\circ}\text{K}$
Void	-0.095%	$\delta k/\%$ void overall
	-0.147%	$\delta k/\%$ void in core only
Lifetime	71 μsec	fresh core
	140 μsec	burnt out core
Increase core volume by 20%	$\delta k = 3.04\%$	
(Fill in corners to make a square core)		

*The answer could also have been obtained by solving a problem in which a line source of neutrons is placed at the center of the reactor and the adjoint fluxes determined when ν was set equal to zero outside the center. An economical method for solving this problem was not immediately at hand at the time, however, so the first method was used.

7. Shutdown Recovery - Xenon Override

When the power of a reactor is reduced, the absorption of xenon is immediately reduced but the decay of its precursor is not affected immediately. This causes the xenon concentration to increase over its equilibrium concentration until its own decay becomes dominant. In AARR, with its high power density, this xenon buildup has an appreciable reactivity effect which cannot be compensated for by rod removal at all points in the core lifetime. If it is desired to regain full power, this must be done immediately after the power setback, before the xenon has reached its peak value, for otherwise it takes more than a day for the xenon to decay away. The time available for regaining full power (setback time) has been determined for power setbacks to 20% and 50% of full power for several points in the core lifetime.

The results of the Candle analysis (see Section 4) have been used to determine the reactivity worth at a given time of a given absorption area. The available k_{ex} can be read from Table 20. The xenon equations were solved on the ANL analogue computer^{12,13} to yield the xenon concentration as a function of time. In this way, the time at which the xenon has used up all the k_{ex} and thus (approximately) the time by which full power must be resumed is known.

Table 22 gives the setback times for 6, 5, and 4% k in experiments, as well as the excess reactivity δk_M that must be present if the setback time is to be effectively infinite.

Table 22

SETBACK TIMES

Operation Time (sec)	Power after Setback (Mw)	Setback Times (sec)			δk_M (%)
		t_1	t_2	t_3	
2×10^5	20	0	70	105	11.3
	50	36	150	260	2.8
4×10^5	20	0	70	105	10
	50	36	150	∞	2.7
6×10^5	20	36	55	70	10.4
	50	36	150	∞	2.8
8×10^5	20	0	0	70	9.6
	50	0	70	∞	2.95
10×10^5	20	0	0	0	9.5
	50	0	70	180	2.87

t_1 = setback time with 6% k_{ex} in experiments

t_2 = setback time with 5% k_{ex} in experiments

t_3 = setback time with 4% k_{ex} in experiments.

Table 22 demonstrates the importance of determining the excess reactivity present in the reactor as a function of burnup. Also, a setback to 20 Mw (from 100 Mw) is almost always certain to result in a complete shutdown, but a setback to 50 Mw is clearly feasible, and if reactivity can be inserted by moving in 12 additional fuel boxes at 50 Mw in about 1 min, the last entry in Table 22 shows that a 50% setback can always be maintained. Recovery from a complete shutdown, however, will nearly always take a minimum of 39 hr.

8. Storage Properties of the Fresh Core

The reactivity of various parts of the fresh core totally submerged in water was obtained for a variety of configurations. This information is summarized in Table 23.

Table 23

REACTIVITY OF FRACTIONAL CORES STORED IN WATER

<u>Core Fraction</u>	<u>Control Rods In</u>	<u>Control Rods Out</u>
1.0	0.847	0.931
0.5	0.680	0.724
0.25	-	0.521

(A fractional core is made by cutting parallel to the vertical axis)

9. Power Flattening

Flattening power can substantially increase the peak thermal flux because, in general, it is the maximum power density that limits performance. As can be concluded from the remarks in Section 5, it is of greatest benefit to flatten the flux in the innermost third of the core since most (70%) of the neutrons thermalized at the center of the reactor come from that part of the core.

Methods of flux flattening include: (1) variation of fuel content in the horizontal plane and in the vertical plane, with or without poison variation; and (2) insertion of a sleeve black to thermal neutrons around the thermal column and variation of fuel density.

Flux plots show that the thermal neutron current into the core is large. The black sleeve around the internal thermal column would remove this current. In addition, it would heavily depress the power just

where the fission source is most effective in producing neutrons which will become thermal at the center; to counteract this, the fuel loading must be increased around the sleeve. But this accentuates the peak in the thermal flux in the core which now occurs at the outer edge of the core. In this case, the fuel loading at the outer edge must be decreased. As this problem is pursued, it becomes apparent that the result is not much different in principle from that obtained by method 1.

The difficulty with fuel and poison variation is that the thermal neutron relaxation length is of the order of 2 to 3 cm (in the fresh hot core, it is 2.29 cm) so that of the order of 6 to 9 horizontal zones of fuel concentration will be required to produce a flat power distribution. This imposes severe problems on fuel box design.

A study of the fuel configuration shown in Fig. 20 gave a maximum-to-average power density of 1.85 (overall). The peak thermal flux at 100 Mw is 2.8×10^{15} neutrons/(cm²)(sec). No axial variation in fuel loading was considered in this case. The atomic densities of U²³⁵ and B¹⁰ in the various fuels are given in Table 24. The initial reactivity of this system is $k_{eff} = 0.999$ with the rods in; approximately 1.5%k is lost by flattening the flux.

No burnup studies were made of this core; nor was any attempt made to rationalize the fuel pattern with fuel box design requirements.

FIG. 20
FUEL PATTERN FOR FLATTENED POWER DISTRIBUTION
(SEE ALSO TABLE 24)

Table 24

U²³⁵ AND B¹⁰ CONTENT OF FUELS USED TO FLATTEN POWER

<u>Fuel Type</u>	<u>U²³⁵ (atoms/cm³)</u>	<u>B¹⁰ (atoms/cm³)</u>
1	0.13543 x 10 ²¹	0.32472 x 10 ¹⁹
2	0.17144 x 10 ²¹	0.411069 x 10 ¹⁹
3	0.21084 x 10 ²¹	0.50555 x 10 ¹⁹
4	0.24441 x 10 ²¹	0.58603 x 10 ¹⁹
5	0.27851 x 10 ²¹	0.66780 x 10 ¹⁹
6	0.20487 x 10 ²¹	0.49123 x 10 ¹⁹

10. Stainless Steel Cores

Some preliminary investigations were made of the effect of substituting stainless steel for aluminum. The chief effect is to reduce the reactivity worth of excess U²³⁵, so that additional burnup may be obtained without a corresponding increase in the amount of reactivity to be controlled.

Using the same techniques as with the aluminum cores, a stainless steel core with a uniform loading of 20 kg of U²³⁵ was studied. The initial clean, cold reactivity was 1.135, the Xe¹³⁵ worth was 2%k, and the reactivity life was 45 to 55 days, depending on whether 6% or 4% k_{ex} is assumed present in experiments.

The peak thermal flux per unit power is 4.2 x 10¹³ neutrons/(cm²)(sec) per megawatt. The maximum-to-average power density is 2.4 overall. (These numbers apply to beginning of life.)

After burning for 45 days, the mass of U²³⁵ is reduced to 14.5 kg; this is equivalent to burning about 2.6% of the atoms in the meat. If we assume that it is possible to burn as much as 10% of the meat atoms and to use cermet techniques to get high weight percentages of UO₂ in a stainless steel meat, then it is reasonable to envisage cores with uranium inventories of about 80 to 100 kg and lifetimes of 180 to 240 days.

In general, in the stainless steel cores, the spectral variation at the core reflector interface is large and rapid so that the simple recipes used for analysis of the aluminum core cannot be expected to apply with any accuracy to the analysis of the stainless steel cores. Moreover, at the high U:H atom ratios the fast spectrum deviates considerably from 1/E, and special care must be used in analysis of epithermal effects.

An attempt was made to use a 28-group cross section set with scattering down in energy through all groups and scattering up from the lower 18 groups through as many as 10 groups. A transport theory code was used in an attempt to find the spatial variation of the spectrum, but the problem did not converge after many hours of machine time and was dropped for reasons of economy and shortage of time.

Infinite-medium spectra in mixtures of H and U were studied by use of hand methods, assuming there were ten fast groups and 1 thermal group. The cross sections used, group structure, group fluxes, and average values of η are given in Table 25 for three cases: $N_H:N_U = 20, 30, 40$. The flux was assumed to vary as $1/E$ in each group, and no capture took place in hydrogen.

Table 25
PHYSICS CONSTANTS FOR A STAINLESS STEEL CORE

Group Index j	Lower Energy Limit E_j (ev)	$\sigma_{c+f,j}^*$ for U^{235} (b)	$\nu\sigma_{f,j}^*$ for U^{235} (b)	$\sigma_{er,j}^*$ for H (b)	Group Fluxes, ϕ_j		
					$N_H:N_U = 20$	$N_H:N_U = 30$	$N_H:N_U = 40$
1	1234	7.07	12.2	20.2	1557	1565	1570
2	454	9.97	16.7	20.2	1530	1450	1554
3	320	14.8	25.1	57.9	249.3	242.2	253.5
4	225	21.2	34.1	57.9	248.5	242.3	253.8
5	167	26.7	43.4	67.3	183.7	179.4	188.2
6	32.3	44.4	73.6	13.5	25.34	25.79	27.64
7	10.7	68.2	114.5	16.2	1502	1603	1761
8	3.0	76.2	97.4	16.2	1410	1569	1762
9	1.0	31.8	51.1	20.4	1025	1144	1287
10	0.4	86.5	176.2	21.3	759.7	888.9	1025
11	0	203.5	428.9	-	3036	5202	7900
					$\eta = 1.902$	1.955	1.986
					$\eta = \frac{\sum_j \nu \sigma_{f,j} \phi_j}{\sum_j \sigma_{c+f,j} \phi_j}$		

*These cross sections are averaged over each group under the assumption that the flux is proportional to E^{-1} there. In group 11, SOFOCATE results are used.

σ_{er} - cross section for elastic removal
Infinite Medium Spectra H:U - 20:30:40

From the data given in Table 25 it can be seen that for low values of $N_H:N_U$ the burnup of uranium should, at high burnups, tend to soften the spectrum enough to reduce materially the reactivity control necessary to allow for burnup.

The xenon-135 effect has not been evaluated for these mixtures, but it is safe to assume it to be much smaller than in the aluminum cores, and possibly the xenon coefficient at power might be made slightly positive.

11. Future Work

Future physics work should center around the following areas:

- a. power flattening in the aluminum core;
- b. methods of recovery from fast setback and complete scrams;
- c. study of the economic merits of various control and recovery schemes as a function of core lifetime and control system speed and reliability;
- d. performance of detailed studies of spatial variation of thermal spectra in aluminum cores;
- e. performance of detailed studies of stainless steel cores.

VIII. HEAT TRANSFER

The reactor physics parameters for AARR define the average core power density and gross distribution necessary to obtain the target neutron flux. The principal objectives of the heat transfer studies were then to prescribe a fuel element structure and coolant conditions to cope with the physics dictates plus those additional engineering factors which increase the heat removal needs in a practical system.

A parallel-plate fuel box configuration was selected as a starting point, simply because the fabrication technology is well established and relatively inexpensive. From the physics calculations and critical experiments on a mockup of the AARR flux trap core, the average power density in the core was estimated to be 1.25 Mw/liter for an unperturbed neutron flux of approximately 5×10^{15} n/(cm²)(sec) in the internal thermal column. Further, maximum-to-average flux distribution was measured as 2.73. Allowances made for hot channel and hot spot factors increased the maximum power density for design purposes to slightly more than 6 Mw/liter. As an additional boundary condition, the metal-to-water ratio throughout the fuel region was fixed at one by physics considerations.

With the power density established, answers to five major heat transfer and flow questions concerned with the following were initially sought:

- (1) temperature of the fuel plates;
- (2) burnout at the fuel plate surface;
- (3) failure of the fuel plates under high coolant flows;
- (4) two-phase flow stability; and
- (5) cooling under emergency and shutdown conditions.

Given variables were the plate dimensions, coolant velocity and pressure, and other parameters of the coolant system.

Unfortunately, the answers to some of the questions must be derived from empirical data, and experimental information pertinent to the AARR were not found in the literature. Consequently, a program of exploratory experiments was started to guide the analysis of core heat transfer. The design specifications that are the result of calculations plus preliminary measurements are listed below. Discussion of some of the individual items, including limitations and areas that need additional experimental support, follow section A.

A. Design Parameters

1. Fuel Heat Removal

Design parameters for the fuel region heat removal including the hot channel factors are given in Table 26.

Table 26

HEAT REMOVAL FROM THE FUEL REGION

	100 Mw	150 Mw
Total heat output, Mw	100	150
Coolant flow		
Total flow rate ($\rho = 61 \text{ lb/ft}^3$), ft^3/sec	36.8 (1042 liters/sec)	36.8
Flow area, ft^2	0.92 (855 cm^2)	0.92
Average velocity, ft/sec	40 (122 cm/sec)	40
Pressure		
System (nominal, core inlet plenum), psia	300 (20.4 atm)	450 (30.6 atm)
Total drop across reactor vessel, psi*	74 (5.03 atm)	74
Total drop across core, psi*	70 (4.76 atm)	70
Heat Transfer		
Total core active surface, ft^2	691 (64.2 m^2)	691
Average heat flux, $\text{Btu}/(\text{ft}^2)(\text{hr})$	0.495×10^6 (1.561 $\times 10^2$)	0.743×10^6 (2.34 $\times 10^2$ watts/ cm^2)
Maximum heat flux, $\text{Btu}/(\text{ft}^2)(\text{hr})$	1.743×10^6 (5.50 $\times 10^2$)	2.615×10^6 (8.25 $\times 10^2$ watts/ cm^2)
Burnout flux, $\text{Btu}/(\text{ft}^2)(\text{hr})$	5.1×10^6 (1.6 $\times 10^3$)	4.6×10^6 (1.45 $\times 10^3$ watts/ cm^2)
Average film coefficient, $\text{Btu}/(\text{ft}^2)(\text{hr})(^\circ\text{F})$	12,800 [7.27]	13,400 7.61 watts/ $(\text{cm}^2)(^\circ\text{C})$]
Average power density, Mw/liter	1.29	1.94
Maximum hot spot power density, Mw/liter	4.54	6.8
Temperature, $^\circ\text{F}$		
Average temperature of coolant in fuel region	156 (69 $^\circ\text{C}$)	166 (74 $^\circ\text{C}$)
Coolant temperature rise across fuel (average)	41 (22 $^\circ\text{C}$)	61.5 (34 $^\circ\text{C}$)
Average film drop	38.7 (21 $^\circ\text{C}$)	55.5 (30 $^\circ\text{C}$)
Average fuel center temperature (clean) (7 $^\circ\text{F}$ and 10.5 $^\circ\text{F}$ through fuel)	201 (94 $^\circ\text{C}$)	232 (111 $^\circ\text{C}$)
Estimated corrosion film temperature drop	50 $^\circ\text{F}$ (27.8 $^\circ\text{C}$)	75 $^\circ\text{F}$ (41.6 $^\circ\text{C}$)
Maximum fuel surface temperature (at 3 in downstream of centerline, scale drop included temperature of bulk fluid at this portion 218 $^\circ\text{F}$ - 100 Mw, 259 $^\circ\text{F}$ - 150 Mw)	410 (210 $^\circ\text{C}$)	520 (271 $^\circ\text{C}$)
Maximum fuel center temperature (scale drop included)	433 (222 $^\circ\text{C}$)	554 (290 $^\circ\text{C}$)
Coolant outlet temperature from hot channel	251 (122 $^\circ\text{C}$)	309 (159 $^\circ\text{C}$)
Coolant inlet temperature to reactor vessel and fuel region	135 (57.2 $^\circ\text{C}$)	135
Local hot channel factors		
Flux		
Radial max/avg	2.10	
Axial max/avg	1.30	
Total max/avg	2.73	
Calculation uncertainty	1.05	
Hot channel		
Control skewing	1.10	
Power uncertainty	1.03	
Channel area	1.05	
Channel coolant velocity	1.05	
Fuel content/plate	$\frac{1.02}{1.23}$	

*These items are the estimates for clean, smooth surfaces with no influence indicated for average water density change between the two power levels

Table 26 (Cont'd)

		<u>100 Mw</u>	<u>150 Mw</u>
Hot spot			
Control skewing	1 10		
Power uncertainty	1 03		
Local fuel content	1 03		
Local flux peak	1 06		
Allow thickness	1 10		
Gamma heat loss	<u>0 90</u>		
	1 23		
Heat transfer			
Heat transfer coefficient	1 25		
Channel coolant velocity	1 05		
Channel area	<u>1 05</u>		
	1 38		
Hot Channel Factor = (2 10)(1 05)(1 28) = 2 82			
Hot Spot Factor = (2 10)(1 30)(1 05)(1 23) = 3 52			
Heat Transfer Factor = (3 52)(1 38) = 4 86			
Power densities, Mw/ℓ			
Channel		3 77	5 45
Spot		4 54	6 80
Heat transfer		6 30	9 45

2. Reflector Heat Removal

The reflector heat removal was calculated for an operating power of 150 Mw. The design was found adequate for this level and thus would be more than adequate for operation at 100 Mw. Reflector design parameters are listed in Table 27

Table 27

HEAT REMOVAL FROM THE REFLECTOR

Total heat output, Mw	5 7
Coolant flow	
Total flow rate, ft ³ /sec	4 2 (118 liters/sec)
Maximum velocity, ft/sec	60 (20 9 m/sec)
Pressure	
Total pressure drop across reflector, psi	70 (4 76 atm)
Heat transfer	
Maximum heat flux, Btu/(ft ²)(hr)	1 14 x 10 ⁶ (3 60 x 10 ² watts/cm ²)
Burnout flux, Btu/(ft ²)(hr)	6 3 x 10 ⁶ (2 0 x 10 ³ watts/cm ²)
Average film coefficient, Btu/(ft ²)(hr)(°F)	13,100 (7 43 watts/cm ² -°C)
Temperature, °F	
Average temperature of coolant in reflector region	146 (63 3°C)
Average coolant temperature rise across reflector	21 (11 7°C)
Hot region film drop	87 (48 3°C)
Coolant inlet temperature	135 (57 2°C)
Coolant outlet temperature	156 (66 9°C)

Specific hot channel factors for the reflector have not been established. The preceding values are based on the first row of reflector elements, on the assumption of a conservative heat generation rate of

97 watts/cm³. These reflector pieces are to be the same in cross-sectional area as the fuel boxes. Four $\frac{5}{16}$ -in. cooling holes are centered in the quadrants of each piece. The estimates are further based on the average conditions occurring at the axial midplane.

3. Shutdown Core Heat Removal

a. Normal Shutdown Requirements

It is tentatively established that the requirements for cooling beryllium will determine the condition for shutdown coolant flow in order to obtain the proper flow distribution between fuel and reflector. The ratio of power generation in the reflector immediately following shutdown to the operating power level is approximately 0.24. Since, however, the temperature rise of the coolant may be increased during shutdown, and since a 90% drop in flow rate lowers the value of the heat transfer coefficient by 80%, it appears that 10% of normal flow during the early shutdown period will be adequate. Further investigation of thermal stress in the beryllium will be needed before a final value for shutdown flow is specified.

b. Emergency Shutdown

Emergency shutdown requires extremely rapid rod insertion, since the currently estimated half-time for coastdown is only of the order of 0.6 sec.

c. After Heat Removal With No Forced Flow

The removal of heat without forced flow cannot be accomplished in a reasonable time without coolant boiling. Based upon an extrapolated correlation (see pages 94, 95) to our region of interest, it would appear that a maximum residual surface heat flux of 20,000 Btu/(ft²)(hr) would be safe at the time of fuel subassembly removal.

B. Critical Heat Flux

1. Forced Convection

Several of the existing correlations were used to estimate the critical heat flux for forced and free convection cooling of the AARR parallel-plate fuel elements. Each of the correlations has individual characteristics that limit its validity; furthermore, all have some common reservations to their use in our region of interest. These common reservations are:

- a. The experimental work was performed on the basis of a uniform heat flux distribution.
- b. The experimental work was performed with a single channel, thereby providing little in the way of information indicating the extent to which the burnout flux is coupled to the number of parallel channels.
- c. The experimental work was performed for steady state conditions which do not always exist in a reactor.
- d. Most of the reported work used cylindrical tubes.

In Table 28 burnout heat fluxes are listed to show the spread of values the various equations give when used with data specific to forced convection in AARR.

Table 28

BURNOUT HEAT FLUXES

Source of Correlation Equation Used	Burnout Heat Flux, Btu/(ft ²)(hr)	Comments
Extended method of Bernath, <u>A Theory of Local Boiling Burnout and Its Applications to Existing Data</u> , Third National Heat Transfer Conf , August 1959	12.6×10^6 (3980 watts/cm ²)	Pressure range above 500 psia (34 atm)
McAdams, <u>Heat Transmission</u> , McGraw-Hill Book Co , Inc Third Edition	5.2×10^6 (1640 watts/cm ²)	
Menegus, <u>Burnout of Heating Surfaces in Water</u> , DP-363 (March 1959)	12.5×10^6 (3950 watts/cm ²)	Correlation of 184 points of other investigations
Zenkevitch-Zubbotin, J. Nuc Energy, Part B, Vol 1 (1959).	7.1×10^6 (2240 watts/cm ²)	Pressure range 2000-3000 psia (136-204 atm)
Weatherhead, <u>AARR Heat Transfer at ANL</u> .	4.5×10^6 (1420 watts/cm ²)	Pressure 200 psia Bulk enthalpy 325 Btu

With the area provided by 40-mil-thick fuel plates, the maximum heat flux calculated for 150% of design power is 2.6×10^6 Btu/(ft²)(hr). Acceptance as realistic for AARR of the smallest of the above values for burnout would probably give an adequate safety factor. Nevertheless, it is worthwhile to know with better accuracy the available burnout margin so that more flexibility can be exercised in other areas of the reactor design. To screen these correlations on a rational basis is difficult, because none match exactly the AARR conditions, which are:

Coolant velocity	40 ft/sec (1220 cm/sec)
Hydraulic diameter	0.08 in. (0.203 cm)
	Plate spacing 0.04 in. (0.102 cm)

Coolant liquid density	61 lb/ft ³ (0.975 gm/cm ³)
Coolant vapor density	0.53 lb/ft ³ (0.975 gm/cm ³)
Nominal pressure	~260 psia (17.7 atm)
T _{sat} - T _{bulk}	235°F (130.5°C)

In order to come to a tentative conclusion pending the outcome of a specific testing program, the various correlations were examined in more detail.

The statistical study by Menegus (see Table 28 for reference) describes the 95% confidence level as requiring a factor of 0.71, which would reduce the tabulated burnout flux to 8.8×10^6 Btu/(ft²)(hr). The indicated factor, however, is only a result of data scatter, and it cannot necessarily be used to make the equation valid for a particular case near the limits or outside the range of the experiments covered.

If, however, the statistical factor of 0.71 for the 95% confidence level based upon the Menegus study of 184 points of other investigators is applied to the Zenkevitch-Zubbotin value, a burnout flux of 5.04×10^6 Btu/(ft²)(hr) is obtained. It might be noted that the McAdams value obtained with surface boiling in an annulus falls within the statistical limits of this correlation.

Experimental work was performed at ANL with a 0.045-in.-diameter tube of L/D ratio of 100. Equipment limitations restricted the burnout tests to values of subcooling much lower than those anticipated in the AARR (30°F vs 235°F). Since burnout flux normally increases with subcooling, the values obtained from these experiments are low, and a burnout point greater than the value of 4.5×10^6 Btu/(ft²)(hr) should be expected. Thus, there is reason to believe that the ANL experiments also fall within the range of the Zenkevitch-Zubbotin correlation adjusted by the Menegus statistical factor of 0.71.

It must be emphasized that additional experimental work should be done to include the factors of (a) coupling between channels; (b) particular geometric factors associated with fuel plate assembly geometry and local details; (c) nonuniform flux distribution; and (d) transient effects to account for the possibility of burnout during such an accident as would cause large flow variations. The Zenkevitch-Zubbotin correlation is used as a design guidepost now because it is consistent with experimental data that gives lower burnout values. This, possibly conservative, choice must be accepted until the influence of the factors listed previously is known quantitatively.

2. Free Convection

For burnout during free convection, such as after shutdown, information as to permissible heat flux levels in a parallel-plate assembly is meager. Here again the Menegus correlation was tried, but a large

extrapolation is required and the validity is questionable. This correlation when used for zero-velocity, saturated water at 14.7 psia gives a value for Q'' of $1.4 \times 10^5 \text{ Btu}/(\text{ft}^2)(\text{hr})$. Data from other sources for "unrestricted" flow give a value for Q'' of $3.6 \times 10^5 \text{ Btu}/(\text{ft}^2)(\text{hr})$ in the case of horizontal wires or flat surfaces. If we refer to the results of the Menegus correlation and use his statistics for a 99% confidence level, we would obtain $0.45 \times 140,000 = 63,000 \text{ Btu}/(\text{ft}^2)(\text{hr})$. Again, considering that this correlation emphasized the use of data from single-tube tests with uniform heat flux distribution, it is not unreasonable to insist that the permissible average heat flux be less than the preceding value.

Some measurements of free convection burnout were made at ANL, but several problems arose: plate buckling, burnout at electrical connections, and flow chugging. Nevertheless, the plate disposition approximated that proposed for the AARR, and the lowest average burnout flux value reported for channels with open sides was $23,700 \text{ Btu}/(\text{ft}^2)(\text{hr})$ with 0.035-in. channels, and $41,000 \text{ Btu}/(\text{ft}^2)(\text{hr})$ for 0.060-in. channels. These average values occurred over most of the plate since electrical heating was used. An experimental analysis indicated that a maximum-to-average flux ratio of approximately 10 occurred with a peak at the electrical contact where sharp gradients existed.

The ANL experiments show that a high peak flux may be tolerated in the ideal situation; a sharp reduction occurs when the channel spacing is decreased. Since some plate distortion can occur during operation or afterwards (by accident) during fuel transfer to the canal, the more conservative value of $20,000 \text{ Btu}/(\text{ft}^2)(\text{hr})$ is used to establish the decay cooling time. A second reason for this choice is to provide extra margin of safety while the fuel is in the canal, because here one containment barrier is lost (the pressure vessel).

3. Delay Time for Convective Cooling

If the heat transfer factor is neglected, the local hot spot - hot channel factor of 3.52 applied to the average heat flux of $0.495 \times 10^6 \text{ Btu}/(\text{ft}^2)(\text{hr})$ yields a maximum heat flux of approximately $1.75 \times 10^6 \text{ Btu}/(\text{ft}^2)(\text{hr})$. Under the assumption of a permissible Q'' of $20,000 \text{ Btu}/(\text{ft}^2)(\text{hr})$, the delay time before fuel element removal is estimated. Let P = power at removal time and P_0 = local maximum power under normal operation, and let $P/P_0 \approx 0.07 t^{-0.2}$, where t is delay time in seconds after shutdown. Then,

$$\frac{P}{P_0} \approx 0.07 t^{-0.2} = \frac{20,000}{1,750,000} = 1.14 \times 10^{-2}$$

Therefore,

$$t \approx 8700 \text{ sec} = 2.42 \text{ hr}$$

It is reasonable, therefore, to expect that a delay time of about 3 hr will be sufficient under the postulated conditions.

C. Upflow Versus Downflow

The advantages of downflow start with the elimination of a top hold-down structure. This speeds refueling and makes it unnecessary to compensate for axial fuel expansion. Also, coolant flow in the direction of rod scram provides a secondary source of power assist to gravity drop. Finally, downflow sweeps the N^{16} activity away from the top cover to reduce shield requirements there.

Usually, the objection to downflow stems from the consideration that a reversal of flow is necessary if the core is to be cooled by natural convection following loss of pumping power. To investigate this problem, an analysis was made to see what effect initial flow direction had on flow stability during coastdown. In brief, it was found that the coastdown flow decayed too fast to provide appreciable shutdown cooling in either case. Also, it is necessary to rely on an auxiliary power source to maintain circulation for the first few hours after shutdown because thermal buoyancy effects cannot give adequate flow rates. Consequently, the initial flow direction does not alter the requirements for emergency cooling and downflow was elected for the advantages listed previously.

D. Two-phase Flow Stability

Some theoretical studies of stability of water flow through heated passages have indicated the need to avoid the velocity regions where the pressure drop was not single valued with respect to flow rate. If the method of Chilton (see Appendix B) is used to estimate the "critical" coolant velocity for the AARR core, it is found that the flow conditions specified for heat removal fall within an unstable region. To rectify the situation raised by this particular analysis, three possibilities are suggested:

- (1) Orifice each channel of the fuel boxes.
- (2) Raise the nominal system pressure from 300 to 600 psia.
- (3) Increase velocity.

None of the solutions are particularly compatible with a straightforward reactor design. An orifice for each channel in the fuel elements, where the plate separation is 40 mils, would require extraordinary fabrication techniques. Furthermore, additional restrictions in the flow path would increase the opportunity for plugging. On the other hand, higher system pressures are undesirable also. First, costs go up; second, the stronger beam tubes required reduce neutron transmission.

Work by recent investigators on the stability problem (see Appendix B) shows that it may be quite reasonable to operate with velocities as low as that characteristic of the bottom of the trough of the curve of pressure drop versus mass flow rate. Then, the critical velocity for full-power heat output would be in the neighborhood of 20 ft/sec. The specified coolant velocity, 40 ft/sec, for the design thus would provide an adequate margin of safety. Additional experiments will be necessary to confirm these conclusions.

E. Fuel Box Flow Tests

The design of a high-performance fuel element raises considerable incompatibility between the need for large surface area (thin plates) to limit the heat flux to reasonable values and of strength to withstand high flow rates (thick, supported plates). Two principal modes of hydrodynamic failure were considered in order to establish the structural properties sufficient for coolant velocities in the range from 40 to 50 ft/sec. The first is a divergence phenomenon involving hydrodynamic and elastic forces; there is a critical coolant velocity above which any slight disturbance would cause the fuel plate to deform increasingly in one direction until failure occurs. The second failure mechanism is from static loading caused by unequal flow within adjacent channels. In an attempt to overcome this latter problem, side plates were eliminated from the fuel box to encourage equalization of static pressures between channels. This also removes "corners" in the channel which may hurt heat transfer. Figure 21 is a photograph of such a fuel box made from 50-mil-thick aluminum plates.

Fig. 21

Fuel Box before Flow Test

A test loop was used to see what effect the open channels would have on collapse velocity and coolant distribution. The first fuel box tested had 50-mil plates and channels. At an average water velocity of 39 ft/sec, the velocity measured in four individual channels varied from 37.4 to 40.3 ft/sec, to give a 7.4% deviation from the average. (The predicted velocity deviation based on structural tolerances was 6%.) After a slight increase in average flow velocity during the next test run, a sudden change in the pressure drop across the test section was noted. Examination of

the box revealed plate distortion at the inlet end (see Fig. 22). Although the calculated critical velocity for divergence failure was 41.3 ft/sec, the results of the test did not necessarily verify the prediction.

Fig. 22

Inlet of Fuel Box after Flow Test

When the fuel box failed, the upstream coolant temperature was 295°F and the test section outlet pressure was 50 psig. If the inaccuracy of the measurements and any local reduction in static pressure from local increase in velocity are considered, local boiling in the fuel box becomes possible. In this case, a small vapor pocket could cause partial flow blockage and a resulting structural failure of the fuel box.

A second series of tests was started to obtain a better concept of the failure mechanism. This time, however, thinner (30 mil) fuel plates were used so that the predicted failure velocity would be low enough to insure subcooling throughout the test section. Close to the critical velocity (21 ft/sec) predicted for divergence, a periodic deformation of fuel plates was observed through viewing ports in the test section. Thus, the evidence from this test indicates that the failure of the current design

of fuel element is from divergence rather than by simple hydrodynamic loading created by an irregular width of the coolant channel. (A discussion of the method used to predict failure is given in Appendix A.)

One way to increase the critical divergency velocity, if open-sided fuel boxes prove unsatisfactory, is to decrease the unsupported width of the fuel plate with longitudinal ridges. Nevertheless, at this time, aluminum plates thinner than 40 mils cannot be considered a proven method for increasing heat transfer area.

IX. SHIELDING

In this section the shield requirements and construction features are given. Calculational methods used are described in Appendix D. The radiation dose criteria for which the shield was designed are listed in Table 1 of Appendix D.

A. Reactor Radial Shield

The minimum radial distance from the reactor axial centerline to the outer surface of the shield is 4.37 m along the horizontal through holes. Various regions along the radius are listed in Table 29. At the shield face the total flux of gamma energy during reactor operation will be less than $200 \text{ Mev}/(\text{cm}^2)(\text{sec})$ or 0.25 mr/hr . Both the fast and thermal neutron flux drop below $1 \text{ n}/(\text{cm}^2)(\text{sec})$ within the shield; consequently, they are not significant outside. The individual regions of the radial shield are discussed next.

Table 29

OVERALL SHIELD IN THE DIRECTION OF
THE HORIZONTAL THROUGH HOLES

Region	Composition (Vol %)	Density (g/cc)	Radial Thickness (cm)	Radius of Region Outer Face (cm)
Core		1.92	25 (equivalent)	25
	45 H ₂ O	0.97		
	55 Al	2.7		
Reflector		1.76	31 (equivalent)	56
	10 H ₂ O	0.97		
	90 Be	1.85		
Water		0.97	65.92	121.92
Vessel	Steel	7.6	1.91	123.83
Water		0.97	3.81	127.64
Concrete Thermal Shield	Steel	7.6	5.08	132.72
Water			80.24	212.96
Concrete Liner	Steel	7.6	1.27	214.23
Biological Shield	Magnetite	3.52	212.54	426.77
Recessed Space	Air		10.2	436.97

1. Biological Shield

A plan view through the reactor midplane is shown in Fig. 10-A. Between the reactor vessel and the main concrete shield is a water region in which there is a steel thermal shield to reduce the heat generation in the concrete itself. There is, of course, a concrete liner (steel) to separate the water and main biological shield.

The rectangular area around each horizontal experimental facility is recessed into the outside face of the biological shield a distance of 10.2 cm perpendicular to the facility axis. The overall shield was designed to satisfy the dose-rate criterion at the inner surface of this recess, which is the point of minimum shield thickness for the reactor.

In the direction of the spent-fuel transport and storage canal, the thickness of the water zone between the thermal shield and the concrete liner is 86 cm greater than the water zone along the other radii, since this space must accommodate the reactor coolant pipe. The magnetite concrete (at the end wall of canal) need be only 46 cm in thickness, because the canal water suffices for the remainder of the shield.

As for the biological shield in general, the inner radial half of the magnetite concrete extends upward to a height of 1.83 m (see Fig. 12) above the reactor midplane while the outer half of the annulus is stepped 30.5 cm higher. The biological shield is continued in the vertical direction with ordinary concrete ($\rho = 2.35$ g/cc), the outer surface of which is stepped radially inward at a height of 4.72 m (including the steel base thickness of the service elevation) above the core midplane. From this step, the concrete extends upward in annulus geometry a remaining 2.13 m. At a height of 2.35 m above the core midplane, the inside surface of the shield is stepped outward 5.08 cm to accommodate the steel ring which supports the shield above the vessel.

The top annulus of ordinary concrete need be only 46 cm thick for shielding purposes, but it has been given an 80-cm (including the 1.27-cm stainless steel liner) thickness to provide sufficient structural support around the upper extremities of the hollow vertical shafts which house and furnish access to the beam hole gates. The portion of the gate above the level of the beam hole will be a cylinder of magnetite concrete about 60 cm in diameter.

2. Thermal Shield

This shield is a 5.08-cm-thick, steel cylindrical annulus having a height of 2.28 m. It extends a height of 1.07 m above the reactor midplane. For radiation penetrating above the top of the thermal shield, the greater water thickness traversed offers the necessary thermal protection for the concrete.

The thermal shield will be placed as close to the vessel as possible to maintain a minimum steel volume; the shield will, however, be placed beyond the reinforcing pads around each vessel opening to minimize the area of the holes in the shield and radiation streaming.

a. Shield Region Cooling Requirements

Besides cooling the thermal shields, the shield coolant water will also cool the exterior of the vessel, the water region, the concrete liner, and the concrete. It is estimated that a flow rate of about 200 liters/min with a 50°C temperature rise will supply the necessary cooling for these regions. The greatest heat flux is at the top surface of the 11.4-cm thermal shield directly below the vessel, and since this does not exceed 2 watts/cm², high coolant pressure and velocity are not required.

The smallest flow area for the shield coolant water will be along the primary inlet coolant header. This area should be small in order to reduce the radiation streaming. A water gap of 1.27 cm is reasonable with respect to fabrication and placement tolerances.

b. Construction Material

The radial thermal shield also provides support for the reactor pressure vessel, so it must be sufficiently radiation- and corrosion-resistant to last 20 yr. It need not satisfy the requirement of low induced activation, i.e., low manganese, cobalt, and tantalum concentration.

The cheapest material - carbon steel - would require either a protective coating or a corrosion inhibitor in the shield water. The latter is undesirable because of activation and leakage into the canal. Until more information on the behavior of the various possible coating materials is available, stainless or stainless-clad steel is the preferred material for the thermal shield.

3. Reactor Vessel Thermal Shields

The inside diameter of the spherical midsection of the reactor vessel was fixed at 2.44 m in order to allow space between beam holes. This size gives a large water region between vessel and core to attenuate the fast neutron and gamma fluxes. The resultant total gamma-energy flux and the fast neutron removal flux at the vessel wall are 1.4×10^{13} Mev/(cm²)(sec) and 1.7×10^8 n/(cm²)(sec), respectively.

With this spherical vessel design (see Fig. 8), no extensive thermal shielding is required. Local steel shields will circumscribe the vessel penetrations to decrease the thermal stresses in the thicker reinforced regions and to minimize radiation streaming. In addition, some

thermal shielding may be used where the vessel changes from spherical to cylindrical sections. It is quite possible such protection is not necessary, since advanced techniques have been developed to determine the behavior of materials in the elasto-plastic region under the combined effect of mechanical and thermal stresses. A fundamental difference between the mechanical and thermal stresses is that yielding produces relaxation of the thermal stress. The work of experimenters in the field, such as of Langer,¹⁴ has shown that thermal and mechanical stresses do not always act as though they were additive. In some cases, the thermal stress can be completely neglected, even though it may be quite large. This method of stress design has been employed for some components of the EBR-II.

4. Beam Hole Shields

Generally, the beam hole components, such as the collimators, the lower section of the gate or even the entire gate, the internal shielding, and the external shielding, will depend upon the experiment being conducted, and it will be the concern of the experimenter to insure adequate shielding in situations when the original shielding components are not used.

During the time the beam holes are not in use, the beam hole gate will be rotated 90 degrees to close off the radiation beam with the gate shield. Two in-line beam hole shielding plugs, each consisting of an inner cylindrical plug and an outer annulus (inner surface cylindrical, outer surface square), will complement the gate shield. The gate plus the two plugs will attenuate the fast neutrons and gamma rays to an estimated dose rate of 0.3 mrem/hr and 0.1 mr/hr, respectively.^{14a}

The most difficult beam hole to shield is one which faces the core directly, is closest to the core, has the greatest cross-sectional area, and has the shortest overall length. Such a beam hole is numbered HB-7 in Fig. 10. The elliptical end segment of this beam hole has major and minor axes of 15.9 and 5.7 cm OD, a 0.95-cm aluminum wall, and a length of 30 cm. It begins 39 cm from the core center. The beam hole section between the elliptical segment and the beam gate has a length of 1.63 m. The overall length of the beam hole to the recess in the biological shield is 4.11 m. The diameter of the collimated beam may vary with the experiment being conducted, but it will be somewhat less than the inside diameter of the beam hole shell because of beam hole liners or collimators. The beam tube was, conservatively, assumed to have (for purposes of calculating radiation fluxes) a cylindrical cross section with an inside diameter of 10.2 cm.

The beam hole gate is a cylinder having a diameter of 61 cm and a height for the lower segment of 46 cm. The gate section of the beam hole is defined by a pipe which passes diametrically through the gate and

¹⁴Langer, B. F., Design Values for Thermal Stress in Ductile Materials, 1958 ASME-AWS Joint Metals Engineering Conference Paper.

^{14a}Snyder, T. M., Neutron Shielding Materials, KAPL-704 (Feb 12, 1952).

has a nominal diameter of 15.2 cm (10.2 cm for the smaller beam holes). This pipe passes through a larger pipe having a diameter of 20.3 cm (15.2 cm for the smaller beam holes) and is welded to it in crosslike fashion. Throughout the length of this latter pipe are 2.54-cm thick stainless steel disks in parallel array and separated by 1.27-cm layers of water. The beam hole gate shield is comprised of the laminated steel-water region together with a 0.64-cm-thick boral disk at each end of the region. The remaining portion or body of the lower gate segment is constructed of borated magnetite concrete encased in a steel jacket.

For a core which operates at 2 Mw/liter (thermal heat output) it is estimated that the quality and intensity of radiation impinging on the beam gate shield is 2.8×10^{11} fast removal neutrons/(cm²)(sec), 7×10^{12} thermal neutrons/(cm²)(sec), and 2×10^{13} gamma Mev/(cm²)(sec).

Primarily, the gate shield design was chosen so as to afford maximum attenuation of the fast component of the neutron spectrum consistent with cost and space limitations. Stainless steel was chosen for the inelastic neutron scatterer, but copper would be a better material from several standpoints if system design is such that galvanic corrosion is not a problem. The nickel content of stainless steel is assumed to allow a steel-to-water volume ratio of 2, which is above the desirable experimental value of about 1.5 for carbon steel when neutrons with energies somewhat greater than 1 Mev are to be attenuated.^{15,16}

The function of the boral is to attenuate the impinging thermal neutrons so as to preclude the generation of high-energy capture gamma rays in the stainless steel disks. An attenuation by a factor of 100 will decrease the thermal neutron flux below the value where it is in equilibrium with the fast neutron removal flux in the steel-water combination. For this type of shield, the fast removal to slow (<0.4 ev) neutron flux equilibrium ratio as calculated by RE-34 two-group code is nearly 2. It is not necessary to suppress the capture gamma generation throughout the shield, because the reduced intensity of the thermal neutron flux by the boral, coupled with a capture gamma relaxation length which is slightly less than the relaxation length of the ultimate gamma source (i.e. the fast neutron removal flux), suffices to maintain the steel capture gammas at a comparatively low level.

The boral will last 36 yr under the following conditions: (a) the B₄C-Al mixture in the boral contains 50% B₄C; (b) the beam hole gate will be closed during 40% of reactor operating time; and (c) reactor operating time will be 80% of total elapsed time. This figure is based on

¹⁵Kukhtevich and Tsypin, Attenuation of the Neutron-capture Gamma Rays Created in Iron-Water Shields, J. of Nuc. Energy, Part B: Reactor Technology, 1, 59 (April 1959).

¹⁶Wood, D. E., Intermediate Energy Neutron Leakage Through Iron, Nuc. Sci. and Eng., 5 (1) 45 (Jan 1959).

a homogeneous dispersal of the boron and a boral performance factor of 100 for thermal neutron attenuation. It has been observed that B_4C grain size in Alcoa boral is sufficiently small that there is very little difference between the former assumption and that which is practically realized.¹⁷

The gate shield will be symmetric with respect to the through hole. Such a design is beneficial for two reasons. The outermost boral disk of the gate shield in the closed position will serve to lessen the induced radioactivity at the end of the innermost beam hole shielding plug. In addition, the gate motive mechanism is simplified because it always rotates in one direction.

With beam collimation, the assumption was made that the thermal neutron current is equivalent to the thermal neutron flux at the innermost boral disk. Thermal neutron capture in the boral results in a heat flux of 3 watts/cm² (through one side only) of the cross-sectional area directly exposed to the neutron beam.

If heat transfer permits, the boral disks will be affixed to the outside (with respect to the gate) surface of the steel disks at each end of the gate shield. The heat generated in the boral will be transferred through the 2.54-cm steel disk to the water on the opposite side. If bolts are used in addition to or instead of a suitable bonding agent to secure the boral to the steel, they should not penetrate that portion of the boral which is exposed to the direct neutron beam.

Water was selected for the neutron slowing down medium in the gate shield because of radiation heating. No detailed heat transfer calculations have been performed; it is estimated that without special cooling provisions the temperature rise in the gate shield due to radiation heating will be of sufficient magnitude to either burn, char, or melt other common types of neutron slowing down materials, such as masonite, paraffin, and polyethylene. With the water coolant it will be necessary to supply either a manifold air vent system or water degassification to insure that air does not accumulate in the gate shield.

The concrete surrounding the laminated shield and the through hole of the beam hole gate will consist of 70% magnetite ore, 7% colemanite dust, 8% bound mix water, and 15% lumnite cement. The concrete will be cured under conditions favorable to maximum concrete water content and small shrinkage. The water of crystallization held by the magnetite (1-5%) and the colemanite dust (22%) will also add to the total water content. This water will help in slowing down the neutrons which will stream through the clearances and structure of the forward beam hole sections.

¹⁷Akerhielm, F., Transmission of Thermal Neutrons Through Boral, AE-24, Stockholm, Sweden (Aug 1960).

a. Beam Hole Shielding Plugs

The shielding plug adjacent to the beam hole gate will be 67.3 cm in length and will consist of a center cylinder and a circumscribing annulus having a 43-cm sq cross section. The annulus will remain in place during those experiments not requiring the full cavity space, whereas the center cylinder will generally be in place when the beam hole is not being utilized.

A total of 21 alternating disks for the cylinder or rings for the annulus constitute the plug shielding beginning with 3.81-cm carbon steel followed by 2.54-cm high-density masonite (Benelex, $\rho = 1.4 \text{ g/cc}$).

The second shielding plug is also made up of an annulus and center cylinder, both having a length equal to 76.2 cm. The shielding material is magnetite concrete. The outside diameters of both the annulus and the center cylinder of the second shielding plug will be greater than the respective component diameters of the first shielding plug by an amount equivalent to eight times the clearance widths around the components in the first shielding plug.

b. Steel Liner for the Concrete

At each location where a concrete surface is adjacent to water, a stainless steel liner is necessary. This liner may have additional functions, such as thermal shielding and structural support, or it can also supply the forming required during concrete pouring operations.

The horizontal concrete liner below the reactor vessel is to be 2.54 cm thick, since it bears the body loads of the vessel with its contents and surrounding facilities. Liners for shield plugs will be thinner to meet the individual structural requirements.

B. Shielding above the Core

The water shield needed for removal of spent fuel and subsequent transfer and storage in the spent-fuel canal establishes the top of the canal wall at a height of 7.92 m above the main floor of the reactor building. The height of the reactor midplane above the main floor is 1.07 m and the top of the core is 22.5 cm higher. By leaving a distance of 30.5 cm between the canal water surface and the top of the canal wall, a shielding space of 6.33 m is available axially above the top of the operating core.

The major closure at the vessel top (see Fig. 12) is a 15.24-cm-thick cylindrical annulus penetrated by seven vertical thimbles having 2.54- and 3.81-cm ID. On the axial centerline is an opening through which the reactor is refueled. This opening is sealed with a 61 cm by 69 cm minor closure having a 10.2-cm thickness. For shielding it is not

practical to use a minor closure as thick as the major closure, because the many penetrations in the major closure coupled with the structural design external to the vessel make it impossible to derive full shielding benefit from the total thickness of the major closure.

Just above the vessel closure (see Fig. 12) is a shield of magnetite concrete totaling 94 cm in thickness and including a 1.27-cm stainless steel liner on all surfaces. This shield weighs 60,000 kg and is a stepped construction of two large semicylinders with three rectangular blocks, two of which (for purposes of refueling) are removed and placed on the end wall of the spent-fuel transport canal. With the blocks removed, the fuel handler is able to transfer the spent fuel laterally to the deeper portion of the canal without raising the fuel above the concrete shield. Magnetite was chosen for the shield above the vessel because, for equivalent gamma attenuation, it is as cheap or cheaper than all metals.

The remainder of the shield above the core consists of water. At the surface of the water in the spent-fuel removal and transfer pool, the dose rate is 0.25 mr/hr. The percentage contributions to this dose rate are the core - 48, reflector - 31, structure - 20, and N^{16} - 1.

The radiation that penetrates the 2.54-cm vessel neck and the 5.08-cm thermal shield region closure plate traverses an oblique path which is greater than the 10.2-cm thickness of the minor vessel closure (see Fig. 12). This oblique path length is great enough to insure that radiation penetrating the shield along the diagonal will not result in a dose rate significantly different than the dose rate on the axial centerline.

Each of the facilities penetrating the top concrete shield (see Fig. 14) will be surrounded with a stepped stainless steel jacket. These jackets can be removed from their supports to provide sufficient working space for unbolting the facilities from their vessel flanges when it becomes necessary to remove the top concrete shield.

C. Shielding below the Core

The shield integrity below a reactor (see Figs. 12 and 16) is almost always difficult to realize because of leakage at passageways, penetrations and pipes for the shield coolant, control rod drives, core coolant, and flushing systems. Differential expansion of the shield and vessel components also complicates the design.

The shielding arrangement selected is a 2.44-m water shield within the primary inlet coolant header, the bottom vessel closure (not considered in the calculations), and the steel cylinder (40.6 cm in thickness and 1.83 m in diameter). Based on this arrangement, the dose rate at the bottom of the steel cylinder is 7.5 mr/hr at 15 min after reactor shutdown. During

operation the dose rate is 10 r/hr, so that entrance into the control rod room is never permitted during operation.

There are several advantages to this design. It is a clean, simplified, essentially one-piece design. It can be fabricated from carbon steel plate stock or integrally cast in a thicker section to compensate for density differences. It blocks more streaming paths without the help of several other shields than do other designs.

The penetrations provided for control rod drive shafts through the steel cylinder should not weaken the shield appreciably, because the gap widths will be small and, also, the solid shafts constitute a much better gamma shield, because of their length, than does the main shield. Neutron streaming should not occur through the shaft because of the small circumference bounded by a water moderator and also because of stainless steel construction with its nickel content, which increases the iron cross section in the dip regions. There will be an upward flow of water between the control rod drive shaft and its jacket to flush radioactive particles from the control rod drive housings.

It is expected that heat convection to the inlet header and expansion joint plus conduction through the steel shielding cylinder itself will suffice to maintain a satisfactory operating temperature for an impinging gamma energy flux of 0.02 watt/cm^2 during operation.

1. Thermal Shields below the Reactor Vessel

To compensate for the absence of a thick water zone between the reactor vessel and the concrete below the vessel (see Fig. 12), it is necessary to employ a horizontal steel thermal shield which will be complemented with the 2.54-cm concrete liner directly below, the 5.08-cm radial thermal shield, and the thermal shield circumscribing the outlet coolant header as described below.

The horizontal thermal shield below the reactor vessel can be constructed of two steel plates in the form of cylindrical annuli with the following dimensions in centimeters:

	<u>Lower Plate</u>	<u>Upper Plate</u>
Thickness	6.34	5.08
Inner Radius	~65*	~80
Outer Radius	152	122

*Depending on header shield and clearance.

The dimensions of the concrete thermal shield below the outlet coolant pipe are:

Thickness	2.54 cm
Length (at midwidth)	91.0 cm
Width (direction perpendicular to inlet coolant pipe)	trench width

Thus the shield will be rectangular except where its arced side approaches to within 1.27 cm of the annular thermal shield circumscribing the outlet coolant header.

A steel cylindrical annulus of 7.62-cm thickness plus the 1.91-cm header wall and the 1.27-cm concrete liner are required at the upper end of the outlet header to function as a thermal shield for the magnetite concrete in that position. The high costs involved in vessel manufacture render it impractical to increase the thickness of the outlet header wall to the thickness needed for shielding.

Either a stainless steel, two-layer lamination or a steel casting, both having stepped joints, would be the most dependable type shield that could be utilized in this area.

The lip at the top of the annular shield is a stainless steel cylindrical annulus having an outer radius and vertical thickness of 78.7 and 7.62 cm, respectively. With the inner surface of the ring welded to the inlet header, added strength, and perhaps a thinner header wall in a critical stress area, can be realized. Again, coolant passages through the stepped shield should be narrow, preferably not greater than 1.27 cm.

2. Concrete Shield below the Reactor Vessel

The design of the biological shield below the reactor vessel is dependent upon the available space (particularly below the horizontal primary outlet coolant pipe), the shielding arrangement around the primary inlet coolant header, the structural requirements, and the axial centerline shielding in the downward direction as previously described.

The floor of the inlet coolant pipe trench above the control rod drive room has an 85-cm thickness of magnetite. The concrete is, of course, complemented by a 5.08-cm thermal shield and the customary 1.27-cm stainless steel liner. Since it is directly below the steel shield on the axial centerline, this design will limit the dose rate from the core 15 min after shutdown to 7.5 mr/hr.

D. Control Rod Drive Room

The biological shielding required for the wall of the control rod drive room to attenuate the gamma radiation to less than 4 mr/hr is either 90 cm of magnetite or 135 cm of ordinary ($\rho = 2.35$ g/cc) concrete. The dose rate at the outer surface of this wall in the region of the experimental service connections is not expected to exceed 5 mr/hr, except perhaps at the ends of the major streaming paths. Normally, neither personnel nor experimental apparatus will be in this area during reactor operation, since there are ducts through the floor above for line connections.

E. Primary Coolant Pipe Tunnel

An elevation drawing of the primary coolant pipe tunnel and spent-fuel transport canal is depicted in Fig. 2. The tunnel is situated directly below the entire length of the canal so that the ceiling of the tunnel need be only that thickness required for structural purposes.

It will be necessary to provide a steel liner for the lower portion of the tunnel or to use a sealer, such as an epoxy resin, in order to minimize concrete surface contamination intrinsic in radioactive coolant leaks. Further investigation is needed to determine if a ventilation system is required.

Shielding calculations are based on a pipe tunnel of 1.98 m and a width of 1.83 m with a distance of 61 cm between the primary coolant pipe and the tunnel wall.

The axial centerline of the upper 61-cm pipe, which contains coolant flowing to the reactor, is above the level of the main floor in the reactor building. Thus the radiation from this pipe must be attenuated to the value of 0.25 mr/hr at the outer surface of the concrete pipe tunnel wall. Because of space limitations in the area where the tunnel and canal emerge from the reactor biological shield and the value of main floor space, it is desirable to construct the pipe tunnel wall of magnetite concrete ($\rho = 3.52$ g/cc).

The outlet primary coolant water at a position in the coolant pipe adjacent to the outer radius of the reactor biological shield has an N^{16} activity of 3.9×10^6 dis/(cc)(sec). Since it takes 5 sec for the coolant leaving the N^{16} production area to reach this position, the N^{16} activity has decayed by a factor of 1.6. Because the coolant traverses the pipe in the tunnel in only 4 to 5 sec, the wall thickness (1.22 m of magnetite) of the entire length of the pipe tunnel was designed on the basis of the above N^{16} activity.

On each side of the primary coolant pipe tunnel the beam hole opening is in close proximity with the outside surface of the pipe tunnel wall. In order to provide adequate working space around these two openings without extending the biological shield, a niche in the shape of a triangular prism will be formed in the wall of the pipe tunnel. The maximum recess depth along the normal to the wall is 40 cm, whereas the height is 2.13 m above the floor level. The complementary shielding at this location will be in the form of a 7.62-cm-thick lead slab bounded by a 1.59-cm layer of steel. The slab will extend 15 cm below the floor level and 7.62 cm above the top of the niche. The width of the slab will be 1.22 m and it will extend perpendicularly outward from the biological shield surface a distance of 61 cm, i.e., radially from the core instead of parallel to the pipe tunnel, so that the slab width is minimized.

F. Spent Fuel

The spent-fuel canal (see Fig. 2) is divided into three distinct sections: the spent-fuel removal and transfer pool, the spent-fuel transport canal, and the spent-fuel storage and handling canal. The transport canal is separated from the other two sections by a weir at each end.

The inside walls of the canal will be lined with stainless steel. Although this type of liner does not have the light reflectivity of the white tile used in the MTR design, it is easier to clean, has greater durability, and should suffer less leakage after long service if the joints are carefully welded and inspected. Underwater lights will provide the proper illumination required for visualization of underwater operations.

1. Spent-fuel Removal and Transfer Pool

The spent-fuel removal and transfer section of the canal consists of a cylindrical pool of water above the reactor. It has one leg extending toward the transport canal so that spent fuel may be quickly transferred laterally from the reactor vessel to the transport cart located in the deeper part of the canal.

The height of the water in the pool is 3.96 m above the top of the shield region closure plate. Since the core has an active height of 45 cm, it should be possible to maintain a 3.35-m water shield above the top of the core during lateral spent fuel transfer. Discounting pool water activity, the dose rate at the water surface from a 2-hr decay core (which has operated for 300 hr at a specific power of 2 Mw/liter) is 1.4 r/hr. The lateral distance between the reactor axial centerline and the drop-off ledge of the pool is about 4.5 m. If it is assumed the spent fuel can be transferred over this distance in a period of 30 sec, the dose received by a fuel handler who either performs or assists in the transfer is 12 mr.

2. Spent-fuel Transport Canal

The spent-fuel transport canal extends between the spent-fuel removal and transfer pool at the edge of the reactor to the spent-fuel storage and handling canal in the heat exchanger building. The main function of this canal is to provide adequate shielding and cooling during the transport of spent fuel to the storage area.

Because the spent-fuel transport will not ordinarily take place while the reactor is in operation, it is unnecessary to provide sufficient canal wall shielding to satisfy the reactor building dose rate criterion of 0.25 mr/hr. For this reason the magnetite concrete wall thickness of the canal is the same as that for the primary coolant pipe tunnel, i.e., 1.22 m. The upper portion of the canal wall can be constructed in a slanted design. The slant begins at a point 4.57 m below the top of the canal wall and terminates at a distance of 91.4 cm from the top of the wall, at which point the wall thickness is 30.5 cm.

The water depth in the transport and storage canal sections is largely determined by the thickness of the water shield deemed necessary during spent-fuel removal. The designs of the canal sections are compatible in that there is sufficient space between the coolant pipe tunnel and the surface of the water shield above the reactor to allow an adequate water layer of 6.1 m in the deeper sections of the canal.

Only 4.88 m of the 6.1-meter water depth in the canal is available as a water shield above the top of the core. The reason for this is that (a), the active height of the core is 45 cm, and (b) the bottom of the core must be separated from the canal floor by 76 cm of water. The bottom shield is necessary to limit thermal stress induced in the canal floor by the gamma radiation from the spent core. In this same respect a spent-core cask or a transport cart core shroud must have the equivalent of a 1.59-cm steel wall because there is only 66 cm of water between the canal wall and the spent core resting on the transport cart. There should be no shielding placed directly above or below the core because it will hinder core cooling by natural circulation.

The width of the transport canal is 1.83 m. At the exterior surface of the 1.22-m magnetite canal wall the dose rate from a 2-hr decay core in its shroud or cask is 10 mr/hr. If it is necessary to store a spent core of short decay in this section of the canal during reactor operation, it may be desirable to supply additional shielding for the core in order to reduce background radiation in the experimental area. It is expected that other various radioactive items which are stored along the sides of this canal will not need supplementary shielding.

Along one side of this canal section there is space for conducting zero-power critical experiments.

3. Spent-fuel Storage and Handling Canal

This section of the canal is located in the heat exchanger building adjacent to the primary coolant heat exchanger cubicle. The dimensions of the water in the canal are:

Length	15.24 m
Width	3.05 m
Height	6.10 m

The walls are constructed of magnetite concrete having the same configuration and thickness as the walls of the transport canal except for the wall common to the heat exchanger cubicle.

The number of cores that will be stored at any one time is mainly dependent upon the proposed AEC regulations pertaining to the shipment of irradiated fuel elements. According to preliminary calculations, a core decay time greater than one year is required. If this is correct and the proposed regulations are binding, then canal storage capacity for 30 to 40 spent cores may be necessary.

It is planned to remove a spent core from the reactor vessel 2 hr after reactor shutdown. At this time a core which has operated for 300 hr at a specific power of 2 Mw/liter will liberate 1.3 Mw of thermal energy as a result of fission product decay. The decay power of the preceding core is only 74 kw or less by about a factor of 17. If the shutdown time is short, there is a slight increase in the total core decay power when the core cycle time is decreased. Since it is not expected there will be other comparable sources of heat liberation in the spent fuel canal, a cooling system having a 2-Mw capacity should prove successful.

4. Shielding and Radiation Dose

As was mentioned above the 2-hr decay core provides nearly all of the heat liberation in the canal. For this reason it is not necessary to consider any other source within the canal in the shielding calculations.

The dose rate criteria for the outside surface of the canal wall and the surface of the canal water are 4 mr/hr and 2 mr/hr, respectively, for a 2-hr decay core. These dose rates are, perhaps, a little higher than it is customary to allow, but it must be remembered that after 1-day core decay the source strength decreases by a factor of 3.4 below that at 2 hr of decay. The dose rate allowance at the water surface is smaller than at the exterior canal wall surface since there is a 2.5 mr/hr localized contribution from the adjacent heat exchanger cubicle during reactor operation. With the 4.88-m water shield above the top of the 2-hr decay core, the dose rate at the water surface is 1.2 mr/hr from the spent core.

Except for the wall common to the primary coolant heat exchanger cubicle the walls of the storage canal have the same cross section as the transport canal wall, i.e., 1.22 m of magnetite concrete at the base. If it is assumed that the core surface is separated from the 1.27 cm canal liner a distance of 76 cm, the storage rack side shield must have the equivalent of a 3.18-cm steel wall. In this particular case the entire 76-cm water shield does not function as a concrete thermal shield but rather as part of the biological shield to reduce the steel thickness of the storage rack.

G. Heat Exchanger Building

A refined analysis of the shielding problems existing in the heat exchanger building (see Figure 11) has not been made. The work that was done has uncovered many of the problems which exist, however, and will serve as a stepping stone to more detailed and accurate calculations in the future.

1. Primary Coolant Heat Exchanger Cubicle

The header at the top of each of the four heat exchangers represents the largest source of radiation which penetrates the ceiling of the heat exchanger cubicle. The N^{16} radiation at the floor level above the cubicle was calculated for a point on the axial centerline of the second on-stream heat exchanger. A magnetite concrete thickness of 1.07 m is required to satisfy the dose rate criterion of 2.5 mr/hr at the floor level. Approximately 20% of this dose rate results from radiation emitted by the heat exchangers adjacent to No. 2.

On the basis of the above calculation it is estimated the magnetite walls of the cubicle will have to be 91.4 cm in thickness in order to attenuate the N^{16} radiation to a dose rate of 75 mr/hr in the adjacent cubicles. In the perpendicular direction to the exterior wall (also 91.4-cm magnetite) there is very little distance attenuation from the tops of the 4 in-line heat exchangers (all other sources are at a level which is below grade). There will be sufficient attenuation by the 61-cm-thick ordinary concrete reactor building walls to reduce the 7.5 mr/hr exiting the cubicle to less than the 0.25-mr/hr criterion within the reactor building. The exception to the 91.4-cm wall is the wall separating the primary coolant heat exchanger cubicle from the spent-fuel storage and handling canal. The dose rate criterion at or above the surface of the canal water for radiation escaping the heat exchanger cubicle is 2.5 mr/hr. Because there is some degree of oblique penetration of the magnetite wall and also because of the water shield in the canal, the necessary wall thickness is only 76 cm. If hydraulic pressures permit, the entire wall on this side of the storage and handling canal may be of the same thickness. With a wall this thin, however, the dose rate in the heat exchanger cubicle or one of the cleanup-system cubicles could be several hundred mr/hr from a 2-hr decay core

within the canal. This indicates the need for regulatory procedure or a shielding cask for short-decay spent fuel which is handled or stored in the canal in a manner which would be a radiation hazard to personnel in adjacent cubicles. A bare 2-hr decay core which is placed in the canal storage rack such that the core surface is 1.83 m from the 76-cm cubicle canal wall results in a dose rate of 60 mr/hr at the inside cubicle wall surface directly opposite the core.

If more detailed calculations indicate the shield thicknesses are strongly dependent upon radiation emitted by the heat exchangers, it may be more economical to supply heat exchanger shadow shielding rather than to design the entire shield for localized requirements.

2. Secondary Coolant System Cubicle

In the secondary coolant system cubicle the dose rate from the primary coolant system cubicle is 2.8 r/hr after a complete core meltdown. This calculation is based upon information obtained from the Westinghouse Test Reactor incident.

The 91.4-cm magnetite concrete wall separating these two cubicles is as thin as any other wall adjacent to the primary coolant system with the exception of the walls separating adjacent cubicles from the primary coolant cleanup system cubicle where the sources are either smaller or have their separate shield. Consequently, the above value is representative of the highest dose rate to which personnel are likely to be exposed in the event of an AARR complete core meltdown that is contained in the primary coolant system.

3. Shield, Canal, and Experimental Cooling and Cleanup Systems

There have been no calculations made to determine the shielding requirements for these cleanup system cubicles because the sources of activity are not yet defined. If the shield and canal coolant systems are combined, the conceivable sources of activity include transmutation of water impurities in the shield region, fission products from defective cores stored and/or being removed from the reactor, and mixing of contaminated primary coolant with the canal water.

Here, as in other cubicles, it will be necessary to determine with respect to operations and economy whether the shielding should be of an individual or multiple unit type or a combination of both. A good design would seem to be one where the concrete walls are adequate to shield the smaller or smallest radiation sources (filters, heat exchanger, or piping) within the cubicle while complementary shielding is utilized around the greater sources of radiation (ion exchange columns).

4. Primary Coolant Degassing, Cooling, and Cleanup System Cubicle

The intrinsic N^{16} activity of the primary coolant circulating through the components within the cleanup system cubicle requires a magnetite shield thickness of about 91 cm to reduce the dose rate to 2.5 mr/hr.

A conservative analysis of the demineralizer system indicates that for an operating power of 155 Mw and a cleanup system flow rate of 260 liters/min the cation unit of AARR requires approximately 84 cm of magnetite for a 2.5-mr/hr dose level. The magnetite shield thickness for the exterior building walls and covers of the primary coolant degassing, cooling, and cleanup system cubicle will be 91 cm, whereas the thickness of walls separating adjacent cubicles will be 84 cm.

H. Spent-fuel Coffin

The spent-fuel decay required before shipment can be made to a reprocessing plant has not yet been estimated. The most important limitations are the AEC proposed regulations pertaining to decay heat removal from solid irradiated fuel during transit outside the confines of authorized use. These regulations stipulate in part that the temperature of the core in its dry, air-filled cask must not exceed a temperature of 82°C below the melting point of the constituent having the lowest melting temperature.¹⁸

Heat transfer calculations have been performed for an AARR core which was in operation for 300 hr. This core was assumed to be suspended in a large room which had walls at a temperature of 38°C. It was discovered that most of the heat was transferred by radiation and that a decay of approximately one year was necessary to preclude the melting of the core. Because less optimum heat transfer conditions exist in an air-filled cask, it will be necessary to increase the spent-fuel decay time and/or rearrange the shipping configuration of subassemblies in order to conform to the AEC proposed regulation.

A calculation was made to determine the cask shielding specifications based on a 60-day decay core. The dose rate criterion was 10 mr/hr at a distance of one meter from the exterior surface of the cask.

It was found that 26.7 cm of lead plus an inner and outer steel jacket, 2.54 cm thick, satisfy the dose rate criterion. The overall dimensions and weight of the cask are listed in Table 30.

¹⁸USAEC, 10 CFR Part 72, Regulations to Protect Against Accidental Criticality and Radiation Exposure in the Shipment of Irradiated Fuel Elements, Section 72.34.

Table 30

SHIPPING CASK SHIELDING DATA FOR
A 60-DAY DECAY CORE

Inside radius	33 cm
Inside height	73.7 cm
Outside radius	62.2 cm
Outside height	1.37 m
Weight	17 metric tons

After one year of decay the radiation intensity is lower by a factor of 10, which is the equivalent of about 4.5 cm of lead.

I. Irradiated Californium Samples

Between core changes the californium sample will be removed and placed in a basket within the reactor vessel. The irradiation will continue until californium is produced in sufficient quantities to merit chemical separation. At proposed flux levels, the total exposure time required is about one year. When the sample is ready for processing it will be removed from the reactor, transported through the canal, and placed in a large shielding cask situated in the storage and handling canal. There is sufficient water in all sections of the canal to provide adequate radiation protection during the operation. This is shown by a calculation made for the Cm^{245} fission product decay radiation penetrating only 2.44 meters of water. After 6 hr decay of an irradiated sample the dose rate is 5 mr/hr. The neutrons emitted as a result of spontaneous fission do not contribute to this dose rate.

Elevation and top views of the Cf^{25} shielding cask are depicted in Fig. 23.

The dimensions of the cylindrical cask chamber are: height - 34.3 cm; diameter - 13.0 cm. A removable annular liner of 1.91-cm- thick stainless steel is provided at the center of the cask in order to accommodate larger samples. The liner is beveled at the top to serve as a guide during sample placement. The cask structural material will be stainless steel to aid decontamination.

The main body of the cask shield will be 60 vol % steel punching scrap and 40 vol % paraffin having a high melting point. The kind of punching steel is unimportant, but the punching heights and diameters should not exceed 2.54 cm in order to preclude excessive radiation channeling and reduced shield thickness resulting from pellet-packing perturbations at the cask boundaries.

FIG. 23
Cf²⁵² TRANSPORT COFFIN

Because steel is more opaque to gammas than is paraffin, the size of the steel pellets affects the gamma attenuation more than the neutron attenuation. The radial midplane shielding thickness of the cask including the liner is 81.3 cm, whereas the outside diameter is 1.76 m.

The stainless steel jacket structure of each of the 2 cask shielding plugs will consist of a 16.8-cm-OD, schedule 40 pipe welded to a cylindrical annulus of smaller OD so as to form a 1.27-cm shielding step midway between the ends of the plug. The maximum clearance between the plug and its shroud will be limited to 0.32 cm to prevent radiation streaming.

The plug shielding will consist of alternating stacked disks of 3.81-cm carbon steel and 2.54-cm polyethylene, consisting per plug of 11 and 12 disks, respectively. The ends of the plug will be capped with 1.91-cm stainless steel disks. Hence, the total shielding length of the plug is 76.2 cm, whereas the shielding height of the cask will be 1.87 m.

The average density of the cask is 5.1 g/cc while the total weight is about 23 metric tons.

No cooling facilities have been incorporated into the design, but the possibility exists that they may be required for a high Cm^{245} concentration and a short Cm^{245} fission product decay time.

The maximum dose rates caused by radiations emanating from the cask surface after 2 yr of continuous sample exposure and a 6-hr decay time are given in Table 31.

Table 31

RADIATION DOSE RATES AT Cf^{252} CASK SURFACE

<u>Radiation</u>	
Fast neutrons	11 mrem/hr
Shield capture gammas	3 mr/hr
Curium ²⁴⁵ fission product gammas	0.3 mr/hr
Aluminum-clad capture gammas	minor
Cf^{252} prompt fission gammas	minor
Curium ²⁴⁵ prompt fission gammas	minor

At a distance of one meter from the cask surface the dose rate totals about 7 mrem/hr.

Appendix A

HYDROELASTIC PROBLEMS IN HIGH-POWER-DENSITY CORE DESIGN

I. INTRODUCTION

Hydroelasticity is a science which studies the mutual interaction between hydrodynamic, elastic, and inertia forces. This interaction can influence the design of high-flux nuclear reactors. Large power densities in the AARR necessitate large heat transfer surface areas with corresponding thin flexible fuel elements and large coolant velocities. Hydroelastic problems would not exist if the core of the nuclear reactor were perfectly rigid. Structural flexibility may not be objectionable by itself as long as the structure is strong enough to withstand the hydraulic loads without impairing performance. Hydroelastic problems arise when structural deformations induce new hydrodynamic forces which in turn produce additional structural deformations inducing still other hydrodynamic forces. Such interactions may tend to converge until a condition of neutral equilibrium is reached, or they may tend to diverge until the structure fails. Hydroelasticity is very similar to aeroelasticity, a science which studies the mutual interaction between aerodynamic, elastic, and inertia forces in the design of airplanes.

There are several possible hydroelastic problems in the design of a reactor core.

One important problem is divergence. Divergence involves only hydrodynamic and elastic forces. Thus inertia and the independent variable time are eliminated from the equation. Divergence implies a coolant velocity above which any slight disturbance would cause the fuel plate to deform increasingly in one direction until failure occurs. This velocity will be called the divergence velocity.

Flutter is another possible problem. In flutter there is a coolant velocity above which any slight disturbance would cause the fuel plate to oscillate with an increasing amplitude until the structure fails. This velocity will be called the flutter velocity. Associated with the flutter velocity is a flutter frequency. The flutter velocity and frequency can also be defined as the lowest coolant velocity at which a fuel element will exhibit sustained, simple harmonic oscillations.

Also possible are failures from buffeting and impulse loads. Buffeting would be caused by flow irregularities caused by the fuel box inlets and support grids. Impulse loads would be caused by major coolant flow changes.

Of course, the fuel plate may also fail under simple hydrodynamic loads. These loads resulting from dimensional variations caused by tolerance requirements can severely modify any hydroelastic calculation.

Thermal stresses also are present as is heat transfer to the fluid. Thus, the problem quickly becomes extremely complicated when considered in detail.

II. DIVERGENCE ANALYSIS

The divergence problem of parallel-plate fuel assemblies has been discussed by many authors.^{1,2,3} To obtain a divergence velocity one writes an equation of equilibrium and solves for the velocity at which the plate deformations go to infinity.

The equation of equilibrium for a flat plate under all the assumptions of classical elasticity is

$$D \nabla^2 (\nabla^2 u) = \Delta p \quad , \quad (A-1)$$

where D is the flexural rigidity of the plate, u is the plate deformation, and Δp is the pressure acting at the plate. ∇^2 is the operator

$$\frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2} \quad .$$

The difficulty of solving Eq. (A-1) depends on the boundary conditions and the form of Δp . Assume Δp is proportional to the deflection and the square of the coolant velocity V , that is,

$$\Delta p \propto \frac{\rho}{h} V^2 u \quad .$$

This assumes a constant mass flow of coolant, that is, $hV = (h + u)(V + \Delta V)$, where h is the original spacing between plates, and that the pressure p is proportional to the dynamic fluid pressure: $p \propto \frac{1}{2} \rho V^2$, where ρ is the coolant density. Then

$$\Delta p \propto \frac{1}{2} \rho (V + \Delta V)^2 - \frac{1}{2} \rho V^2 = \frac{1}{2} \rho V^2 \left[\left(\frac{h}{h + u} \right) - 1 \right] \approx \rho V^2 \left(\frac{u}{h} \right)$$

for small u . Thus,

$$D \nabla^2 (\nabla^2 u) \propto \frac{\rho}{h} V^2 u \quad (A-2)$$

is the equation to be solved for u . The solution will then be examined for the values of V for which u becomes infinite. Consider the plate in the coolant vibrating with a certain natural mode u at frequency ω . The

¹D. R. Miller, Critical Flow Velocities for Collapse of Reactor Parallel-plate Fuel Assemblies, KAPL-1954 (Aug 26, 1958).

²E. B. Johansson, Hydraulic Instability of Reactor Parallel-plate Fuel Assemblies, KAPL-M-ES-9 (Sept 10, 1959).

³G. Rosenberg and C. Youngdahl, Private Communication.

equation of motion for the plate in free vibration can be written as

$$\nabla^2(\nabla^2 u) = k \frac{\partial^2 u}{\partial t^2} = k \omega^2 u \quad . \quad (\text{A-3})$$

The solution for ω in Eq. (A-3) is directly proportional to the square root of the flexural rigidity and inversely proportional to the square of a characteristic plate length, a , and the square root of the apparent mass, k ,

$$\omega \propto \frac{1}{a^2} \sqrt{\frac{D}{k}} \quad . \quad (\text{A-4})$$

Placing Eq. (A-4) in Eq. (A-3) gives

$$D \nabla^2(\nabla^2 u) \propto \frac{D}{a^4} u \quad . \quad (\text{A-5})$$

Assume the plate during divergence takes the shape of the mode u . Then from Eqs. (A-2) and (A-5),

$$\frac{D}{a^4} u \propto \frac{\rho}{h} V^2 u \quad . \quad (\text{A-6})$$

Since at the critical divergent velocity V_d the deformation u goes to infinity,

$$\frac{D}{a^4} u \propto \frac{\rho}{h} V_d^2 \quad (\text{A-7})$$

or

$$V_d \propto \left[\frac{hD}{a} \right]^{1/2} \quad . \quad (\text{A-8})$$

Since the flexural rigidity D is given by

$$D = \frac{Eb^3}{12(1 - \nu^2)} \quad ,$$

where E is the modulus of elasticity, ν is Poisson's ratio, and b is the plate thickness,

$$V_d \propto \left[\frac{hEb^3}{(1 - \nu^2)\rho a} \right]^{1/2} \quad . \quad (\text{A-9})$$

III. SIMPLE HYDRODYNAMIC FAILURE

The equation for simple hydrodynamic failure is similar to the equation for divergence, but failure will now occur when the pressure caused by an initial deformation δ causes the plate to fail.

The equation of deformation, using Eq. (A-2) with a permanent deformation δ rather than a variable deformation u on the right side of the equation, is

$$D \nabla^2 (\nabla^2 u) \propto \frac{h}{\rho} V^2 \delta \quad . \quad (A-10)$$

The theory of plates states that the stress σ is proportional to the bending moment M divided by the square of the plate thickness b :

$$\sigma \propto \frac{M}{b^2} \quad , \quad (A-11)$$

and the bending moment is proportional to the flexural rigidity times a function of the second derivatives of the deformation:

$$M \propto Df \left(\frac{\partial^2 u}{\partial x^2} , \frac{\partial^2 u}{\partial y^2} \right) \quad . \quad (A-12)$$

Thus x and y can be regarded as multiples of a characteristic plate dimension, and so Eqs. (A-10) and (A-12) can be written as

$$D \frac{u}{a^4} \propto \frac{h}{\rho} V^2 \delta \quad (A-13)$$

and

$$M \propto D \frac{u}{a^2} \quad . \quad (A-14)$$

Combination of Eqs. (A-11), (A-13), and (A-14) to eliminate u and M gives

$$\frac{\sigma h^2}{a^2} \propto \frac{h}{\rho} V^2 \delta \quad . \quad (A-15)$$

If the plate fails at a stress $\bar{\sigma}$, then the critical velocity is

$$V_s \propto \left[\frac{h \bar{\sigma} h^2}{\delta \rho a^2} \right]^{1/2} \quad . \quad (A-16)$$

IV. FLUTTER

Flutter is considerably more complex than either divergence or simple hydrodynamic failure.⁴ To gain an appreciation of the problem, a simple illustration will be given. Consider the equation of motion for a simple system of one degree of freedom:

$$M \frac{d^2x}{dt^2} + C \frac{dx}{dt} + Kx = F \left(t, x, \frac{dx}{dt}, V \right) \quad , \quad (A-17)$$

where M is the mass, C is the damping coefficient, K is the spring coefficient, and $F [t, x, (dx/dt), V]$ the forcing function. Note that F is a function of the coolant velocity V , the time t , displacement x , and the derivative of the displacement, dx/dt . We will let

$$F = A(V)x + B(V) \frac{dx}{dt} \quad ,$$

where $A(V)$ and $B(V)$ are functions of the coolant velocity. Assume $x = xe^{-i\omega t}$, substitute into Eq. (A-17), and divide by x to get

$$M\omega^2 + i\omega C + K = A(V) + i\omega B(V) \quad . \quad (A-18)$$

Equation (A-18) can be divided into real and imaginary parts:

$$M\omega^2 + K = A(V) \quad (A-19)$$

$$C = B(V) \quad . \quad (A-20)$$

Flutter will occur for the value of V where $B(V) \geq C$. The frequency ω_f at which flutter will start can be obtained from Eq. (A-19) by substituting the value of velocity V_f which solves Eq. (A-20). The system is then stable as long as the damping is greater than the out-of-phase forcing function. With respect to a plate in the coolant, the plate will be stable as long as the structural and fluid damping are greater than the out-of-phase component of the hydrodynamic forces.

At present it is difficult to imagine any large imaginary hydrodynamic components; thus a detailed flutter analysis has not been performed.

⁴R. L. Bisplinghoff, H. Ashley, and R. L. Halfman, Aeroelasticity, Addison-Wesley Publishing Company, Inc. (1955).

Appendix B

TWO-PHASE FLOW STABILITY IN PARALLEL HEATED PASSAGES
WITH EXTENSIONS TO CURRENT CONCEPTSI. INTRODUCTION

The comparatively recent entry into areas of core design for very high thermal performance may tend to make questions of flow stability require more thorough treatment. This is particularly significant in the case of cores constructed with small rectangular coolant passages as in the AARR. Earlier investigators, such as Chilton,² indicated the need to avoid the velocity regions where the pressure drop was not single valued with respect to flow rate, as shown in Fig. B-1. Here G_c represents the critical

FIG. B-1
PRESSURE DROP VS FLOW RATE
IN SINGLE CHANNEL

flow rate below which the possibility of another, much lower flow rate would be capable (in the boiling region) of maintaining the ΔP equilibrium. Since it has not been uncommon that such calculations^{1,2} have indicated that the required G to avoid this region is in excess of the desired (or required) flow rate for other reasons such as burnout, fuel plate elastic stability, etc., several approaches have been used to examine the problem in

some analytical detail in preparation for experimental work.

With the use of thin, flat fuel plates, the incompatibility between plate thickness requirements for plate stability under hydrodynamic loading and velocity requirements to avoid the two-phase region of flow instability have led to examining the possibility of improving the concept of flow stability in the hope that the unstable region is in fact smaller than previously indicated (see Fig. B-1), enabling operating velocities to be reduced while not sacrificing stability.

A report by Bick³ indicates the feasibility of this approach. The purposes here are to examine this concept and to expand upon it to incorporate the influences of fluid system components in addition to the core itself.

The basic concept used is that of a rather conventional nature in other areas of transient analysis. It primarily states that stability exists

¹ G. Rosenberg, Private Communication.

² H. Chilton, A Theoretical Study of Stability in Water Flow Through Heated Passages, J. Nuclear Energy, 5, 273-284 (1957).

³ J. H. Bick, A New Method for Determining the Stability of Two-Phase Flow in Parallel Heated Channels with Applications to Nuclear Reactors, NAA-SR-4927.

if the system can recover from small departures from a steady state rather than diverge. Bick³ uses certain assumptions which, although appropriate in many areas of stability analysis, require some justification, and this will be treated in the order in which the need for the assumptions arise.

II ANALYSIS AND DISCUSSION

If water be assumed to be the coolant and if a set of n parallel passages is considered through which the water is pumped under steady flow conditions,

$$\Delta p_c = \Delta P_{c1}, \quad 1 \leq i \leq n \quad (B-1)$$

where Δp_c is the true overall pressure drop and ΔP_{c1} is the frictional pressure drop in the i^{th} channel.

A similar condition exists in the entire circuit. Suppose there are m fluid circuit components in series, of which the entire core is one. Under transient conditions in the j^{th} circuit component, the momentum equation is

$$-\rho L_j A_j \frac{dv_j}{dt} = -A_j \Delta p_j + A_j \Delta P_j, \quad 1 \leq j \leq m \quad (B-2)$$

Considering a fluid of constant density, and the first circuit component as the reference, the continuity condition is expressible as

$$v_j = v_1 \frac{A_1}{A_j} \quad (B-3)$$

Using Eq. (B-3), Eq. (B-2) becomes

$$-\rho L_j A_j \frac{A_1}{A_j} \frac{dv_1}{dt} = A_j (-\Delta p_j + \Delta P_j)$$

or, summing over the circuit,

$$-\rho A_1 \frac{dv_1}{dt} \sum_{j=1}^m \frac{L_j}{A_j} = - \sum_{j=1}^m \Delta p_j + \sum_{j=1}^m \Delta P_j \quad (B-4)$$

Since v is proportional to G , it follows that $\sum_{j=1}^m \Delta p_j$ is the total true pressure drop in the circuit or the pressure drop across the pump, and $\sum_{j=1}^m \Delta P_j$ is the total circuit friction drop. Accordingly, Eq. (B-4) can be

³ See page 126 for footnote.

used (by dropping the subscripts) to represent the total system. Thus the momentum equation for the total system becomes

$$\mu \frac{dG}{dt} + \Delta P = \Delta p \quad , \quad (B-5)$$

where μ is a positive constant.

At this point there are several approximating assumptions one could make concerning the nature of the system.

For example, if the flow circuit characteristics are constant and turbulent flow exists, it enables ΔP to be approximated as KG^2 with K constant. This converts Eq. (B-5) to

$$\mu \frac{dG}{dt} + KG^2 = \Delta p \quad . \quad (B-6)$$

Considering small variations δG in flow about the nominal pump operating position, if k is the slope of the pump performance curve at its nominal operating position, the approximation is made that

$$\delta \Delta p = -k \delta G \quad . \quad (B-7)$$

Taking the differential of (B-6) and using (B-7), we obtain

$$\mu \frac{d\delta G}{dt} + (2KG + k) \delta G = 0 \quad . \quad (B-8)$$

If we further assume negligible changes in $\frac{2KG + k}{\mu}$, we obtain (thereby describing small disturbances)

$$\delta G = Ae^{\lambda t} \quad , \quad (B-9)$$

where A represents the initial disturbance given to G , and $\lambda = \frac{-k - 2KG}{\mu}$, with G representing the initial nominal mass flow rate.

From this, the approximate influence of the pump performance characteristics can be seen, as well as the influence of the total mass flow rate in a system of fixed flow characteristic K . Since λ is always negative, stability exists. However, various recovery times would occur depending on the magnitude of λ .

Consider the two-phase pressure drop curve as representing the performance curve of a given channel. In the event that nominal operation occurs in the nonboiling region, the question arises as to whether or not, or under what conditions, a jump to the boiling region can occur should the operating point be in the "unstable" region of Chilton. The conceivable mechanisms might be a sudden change in the operating characteristic K , or a power excursion.

Considering first the change in the characteristic K , the question to be satisfied becomes that of whether or not a sudden increase in K produces an overshoot of whatever the new performance curve position would be. For example, if initial operation under the steady-state condition is implied by K_1 [in the nonboiling region, where $\Delta P = K_1 G^2$, (see Fig. B-2)], a slow change in K from K_1 to K_2 would normally be expected to produce only a new operating curve implied by K_2 in the nonboiling region.

FIG. B-2
PRESSURE DROP VS FLOW RATE IN SINGLE
CHANNEL WITH CHANGE IN K

If this occurs in a set of channels where the number of undisturbed channels greatly exceeds the number of disturbed channels, no significant change in ΔP is expected. The question, then, is that of how G goes from G_1 to G_2 . If no overshoot occurs, there appears to be no danger of "capture" in the boiling region. The following attempts to show that no overshoot or oscillation occurs in this process.

Again using Eq. (B-6), but considering it for a single channel in the core, we assume Δp is held constant but that K increases. If we consider the general case, where $K = K(t)$, Eq. (B-6) becomes the Riccati equation and is converted to a linear second-order differential equation by the substitution

$$G = + \frac{\mu}{Ku} \frac{du}{dt} \quad , \quad (B-10)$$

so that

$$\frac{d^2u}{dt^2} - \frac{1}{K} \frac{dK}{dt} \frac{du}{dt} - \frac{\Delta p}{\mu^2} Ku = 0 \quad . \quad (B-11)$$

If we assume a step change in K , at $t = 0$, we merely use $K = K_2$ for $t > 0$ and $K = K_1$ for $t < 0$, with $\frac{dK}{dt} = 0$:

$$\frac{d^2u}{dt^2} - \frac{\Delta p}{\mu^2} K_2 u = 0 ; \quad t > 0 \quad . \quad (B-12)$$

The solution to Eq. (B-12) is

$$u = A \sinh \frac{\sqrt{\Delta p K_2}}{\mu} t + B \cosh \frac{\sqrt{\Delta p K_2}}{\mu} t \quad , \quad (B-13)$$

or

$$G = \sqrt{\frac{\Delta p}{K_2}} \left[\frac{C \sinh \frac{\sqrt{\Delta p K_2}}{\mu} t + \cosh \frac{\sqrt{\Delta p K_2}}{\mu} t}{\sinh \frac{\sqrt{\Delta p K_2}}{\mu} t + C \cosh \frac{\sqrt{\Delta p K_2}}{\mu} t} \right] \quad . \quad (B-14)$$

Note that $\sqrt{\frac{\Delta p}{K_2}}$ is simply the final steady-state value of G .

Considering the initial condition that $G = G_1$ for $t = 0$,

$$G_1 = \sqrt{\frac{\Delta p}{K_2}} \left[\frac{1}{C} \right] \quad , \quad \text{or } C = \frac{1}{G_1} \sqrt{\frac{\Delta p}{K_2}} = \frac{G_2}{G_1} \quad ,$$

and

$$G = G_2 \left[\frac{\frac{G_2}{G_1} \sinh \frac{\sqrt{\Delta p K_2}}{\mu} t + \cosh \frac{\sqrt{\Delta p K_2}}{\mu} t}{\sinh \frac{\sqrt{\Delta p K_2}}{\mu} t + \frac{G_2}{G_1} \cosh \frac{\sqrt{\Delta p K_2}}{\mu} t} \right] \quad . \quad (B-15)$$

For $G_2 < G_1$, the term in brackets is always greater than unity and approaches unity as the argument of the hyperbolic functions increases. This indicates that a gradual approach to the new steady state occurs. It would, therefore, appear that, with respect to the effect of changes in K , operation close to the bulk boiling region could occur if a reasonable safety factor is provided to cover possible changes in K .

Next, considering the influence of a power excursion, the revised performance curve might look something like the dashed curve in Fig. B-2. If this effect is such as to cause the incipient bulk boiling point to approach the operating position, a jump may be expected to occur. However, the performance curve is normally considered for the worst channel and therefore possible power excursions could be applied to it. In either event, it would seem that approach to the incipient bulk boiling position would be required whether by moving to this position through a power excursion or by approaching its fixed position through increasing K .

In the event that a large number of core passages exist, the analysis of Bick³ further indicates that stability can even exist if one of the passages is operating in the region where $\frac{\partial \Delta P}{\partial G} < 0$. This analysis proceeds by considering that equation 6 holds for all passages, which, written in terms of the i th passage, is

$$\mu_i \frac{dG_i}{dt} = \Delta p - \Delta P_i, \quad 1 \leq i \leq n, \quad (B-16)$$

where ΔP_i is the pressure drop resulting from energy losses in flow.

With the assumptions that the total flow through the core is maintained and that G_i is reduced, then there are corresponding increases in G for the other channels. Considering small changes (a change in Δp , being $\delta \Delta p$, corresponds to δG_i as the change in G_i), it is determined that

$$\mu_i \frac{d\delta G_i}{dt} = \delta \Delta p - \delta \Delta P_i. \quad (B-17)$$

If, based on arguments similar to those used in arriving at Eq. (B-9), a reasonable form for $\delta \Delta p$ and δG_i is assumed to be

$$\left. \begin{aligned} \delta \Delta p &= b e^{\lambda t} \\ \delta \Delta G_i &= a_i e^{\lambda t} \end{aligned} \right\}, \quad (B-18)$$

then Eq. (B-17) becomes

$$\lambda \mu_i a_i e^{\lambda t} = b e^{\lambda t} - \delta \Delta P_i. \quad (B-19)$$

Therefore

$$\mu_i \lambda = \frac{b}{a_i} - \frac{\delta \Delta P_i}{\delta G_i}$$

³ (See page 126) for footnote

or

$$a_i = \frac{b}{\mu_i \lambda + \frac{\partial \Delta P_i}{\partial G_i}} \quad . \quad (B-20)$$

Letting

$$S(\lambda) = \sum_{i=1}^n \frac{b}{\mu_i \lambda + \frac{\partial \Delta P_i}{\partial G_i}} \quad , \quad (B-21)$$

we find, since $\sum_{i=1}^n G_i$ was assumed to be constant, that

$$S(\lambda) = 0 \quad . \quad (B-22)$$

Stability is then determined by the roots of Eq. (B-22) with reference to Eq. (B-18).

Since $\frac{dS(\lambda)}{d\lambda}$ is of constant sign, depending only upon the sign of b , and since there are n poles and $n-1$ roots to Eq. (B-21) all the roots are accounted for by considering their location between the poles. Therefore, if all the $\frac{\partial \Delta P_i}{\partial G_i} > 0$, all the poles are negative and all the roots are negative, and stability exists. Since this holds everywhere in the nonboiling region, an argument in favor of permitting the flow rate (in the limit) to decrease to the trough in the curve of Fig. B-1 or Fig B-2 exists.

If only one of the $\frac{\partial \Delta P_i}{\partial G_i} < 0$, stability can still exist if $b > 0$, provided the following conditions are met. Figure B-3 represents only the single pole greater than zero and the adjacent pole (corresponding to the single channel for which $\frac{\partial \Delta P_i}{\partial G_i} < 0$).

Since $\frac{dS(\lambda)}{d\lambda} < 0$ for $b > 0$, $S(\lambda)$ will look something like Fig. B-3 but may cross the λ axis to the right or left of 0.

From this we can see that for the λ root to be negative (that is, for stability), $S(0)$ must be less than zero. Accordingly, the criterion becomes (for $b > 0$)

$$\sum_{i=1}^n \frac{1}{\frac{\partial \Delta P_i}{\partial G_i}} < 0 \quad \text{if only one of the } \frac{\partial \Delta P_i}{\partial G_i} < 0 \quad . \quad (\text{B-24})$$

In order to show that b must be positive, we need merely consider that for a fixed channel geometry, any flow rejection in one channel

$(\sum G_i = \text{constant})$ is accepted by other channels. Since during this process

$\frac{dG_j}{dt} > 0$ and ΔP_j increases with flow, Δp must also increase, thereby indicating physically that b must be positive. This satisfies the requirements for Eq. (B-24) to hold.

The immediately preceding arguments require that a sufficiently large number of channels be used and that the pressure drop Δp be negligibly small with reference to the characteristics of the operating system. Under these conditions Eq. (B-24) displays the influence of coupling between channels. Should stability just barely exist under Eq. (B-24), the addition of another set of channels, for which $\frac{\partial \Delta P_i}{\partial G_i} > 0$, in parallel, could therefore produce instability. Therefore, the possibility of stability, under this criterion, is reduced with increasing number of channels.

This may pose a problem, since the assumption may no longer be adequate when considering experimental work for which only a few channels, such as 3 or 4, may be used while controlling the power input to only one for examining stability. The conclusions of such experimental work, therefore, may be at variance with experiences on a larger scale.

In order to examine this effect, consider the extension of the preceding concepts to a loop employing only a few parallel channels. A centrifugal pump is to be used which, when looked upon as a "generator" capable of producing a certain pressure difference across itself as a function of its output (flow rate through pump), must have its characteristics imposed upon the system. For example, if we let $\delta \Delta p_s$ be the change in pressure drop in the system other than the core, and if we assume changes in the operating position on the pump performance curve to be sufficiently small, we can write

$$G = G_0 T - \left(\frac{1}{k} \right) (\delta \Delta p_s + \delta \Delta p) \quad , \quad (B-25)$$

where $G_0 T$ is the total initial flow rate. If the loop is mocked up so that $\delta \Delta p_s \ll \delta \Delta p$, we can use the approximation

$$G \doteq G_0 T - \left(\frac{1}{k} \right) (\delta \Delta p) \quad . \quad (B-26)$$

Using the same assumptions as were used through Eq. (B-20) and further noting that, in connection with Eq. (B-22), we used $\sum_{i=1}^n a_i = 0$ because

$\sum_{i=1}^n G_i = \text{constant}$, the only required change, resulting from Eq. (B-26) is that

$$\sum a_i = - \frac{b}{k} \quad . \quad (B-27)$$

Therefore, the new approximate criterion is

$$S(\lambda) = \sum_{i=1}^n \frac{b}{\mu_i \lambda + \frac{\partial \Delta P_i}{\partial G_i}} = - \frac{b}{k} \quad . \quad (B-28)$$

Under these conditions, for one power-controlled channel, in which $\frac{\partial \Delta P}{\partial G} < 0$, the criterion for stability is

$$\sum_{i=1}^n \frac{1}{\frac{\partial \Delta P_i}{\partial G_i}} < - \frac{1}{k} \quad . \quad (B-29)$$

As a result, we can see how stability may be influenced as a function of the slope of the pump operating curve; with all other things being equal, an experiment may reveal whether a stability that simply depends upon the characteristics of the experimental pump exists or not.

Noting the inverse character of the dependence in Eqs. (B-29) and (B-24), we can see that stability is promoted with reduction of $\left| \frac{\partial \Delta P_i}{\partial G_i} \right|$ for the channel with $\frac{\partial \Delta P_i}{\partial G_i} < 0$. This, of course, accounts for producing instability if additional channels are added with $\frac{\partial \Delta P_i}{\partial G_i} > 0$ if the magnitude of $\frac{\partial \Delta P_i}{\partial G_i}$ for the critical channel is not sufficiently small to override the increase.

As a limiting condition, simply let $\frac{\partial \Delta P_i}{\partial G_i} \rightarrow 0$ for the critical channel and note that it merely requires either zero or positive slopes for the remainder of the channels theoretically to insure stability.

It would seem, then, that a simple experiment could be constructed wherein dropping the value of G from some relatively large value would produce a curve similar to that in Fig. B-4. The end point marked x would

FIG. B-4
PRESSURE DROP VS FLOW RATE SHOWING POSSIBLE
LOCATION OF "LIMIT OF OPERATION" WHEN ONLY
ONE CHANNEL HAS $\frac{\partial \Delta P}{\partial G} < 0$

indicate the position of burnout or channel blockage and would be the limit of the experiment, occurring at such a position that $-\frac{\partial \Delta P_i}{\partial G_i}$ becomes sufficiently great that Eq. (B-29) would no longer hold.

The last remarks are specifically intended for an experimental system wherein only a few parallel passages are used with only one being heated to insure positive slopes for all but the heated channel.

Some experimental confirmation of this approach already exists in the work of J. C. Carter.⁴ Here four parallel, electrically heated channels were used and the so-called "limit of operation" appeared to "come at the first minimum of the pressure-velocity curve." Beyond this, the proposed experiment would attempt to see if we could in fact "ride up" the negatively sloped region until the slope magnitude became sufficiently large so as to be incapable of overriding the positive slopes of the other unheated channels.

⁴J. C. Carter, The Effect of Film Boiling, ANL-4766.

Although it is not intended to produce a final experimental design here, it would be desired to include the following:

- (1) Determine pump characteristic curve for various speeds.
- (2) Show stability (or otherwise) for a given heat input to a single channel as it depends upon pump performance.
- (3) For a given "set up" show ability to "ride up" the negatively sloped portion of channel performance curve by controlling either heat input or flow rate in channel.
- (4) By adding parallel channels to a "just stable" system, show dependence of stability on total number of channels when the performance curve for the "hot channel" remains the same.

Appendix C

CONTROL BLADE TEMPERATURE CALCULATIONS

I. INTRODUCTION

The following analysis was performed to determine the effects of gamma heating on the control blade and the surrounding 0.318-cm aluminum sheath. The 0.176-cm-thick cadmium control blade, clad with 0.165-cm aluminum, was assumed 12.1 cm wide for the calculations.

The following conditions were also assumed:

coolant flow channel between sheath and blade	0.102 cm
gamma heating rate (in core)	100 watt/gm
gamma heating rate (above core)	75 watt/gm
pressure drop through blade coolant channel	4.76 atm
inlet coolant temperature	57.2°C

The in-core gamma heating rate was taken to be the same regardless of material or position. It was also assumed that no thermal resistance existed at the surface of contact between the cadmium and the cladding.

II. CONTROL BLADE IN CORE

The total heat generated in the blade and sheath is

$$q_{Tc} = q_b + q_s ,$$

with

$$q_b = q''' (\rho_{cd} V_{cd} + \rho_{al} V_{al}) = q_1(Cd) + q_2(Al)$$

$$q_s = q''' \rho_s V_s ,$$

where

$$q''' = \text{volumetric heat generation (watt/gm)}$$

$$\rho_{cd}, \rho_{al}, \rho_s = \text{density of cadmium, aluminum, and sheath (gm/cm}^3\text{)}$$

$$V_{cd}, V_{al}, V_s = \text{volume of cadmium, aluminum, and sheath (cm}^3\text{)}$$

The total heat was found to be:

$$q_{Tc} = 2.52 \times 10^5 \text{ watts}$$

$$q_1 = 0.917 \times 10^5 \text{ watts}$$

$$q_2 = 0.554 \times 10^5 \text{ watts}$$

$$q_b = 1.47 \times 10^5 \text{ watts}$$

$$q_s = 1.05 \times 10^5 \text{ watts} \quad .$$

The pressure drop through the control channel is the sum of the pressure drop across the blade in the core and the pressure drop in the sheath above the core. Thus,

$$\Delta P = \Delta P_c + \Delta P_{ac} = 5.45 \text{ atm}$$

and

$$\Delta P_c = (K_1 + K_2 + K_3) (V_c^2 / 2g)$$

$$\Delta P_{ac} = (K_4 + K_5) (V_{ac}^2 / 2g) \quad ,$$

where

K_1, K_3, K_5 = entrance and exit head losses

K_2, K_4 = head loss due to friction .

Also

$$K_2 = f l / De_c$$

$$K_4 = f l / De_{ac}$$

where

f = friction factor (assume $f = 0.0185$)

De_c = equivalent diameter in core = 0.203 cm

De_{ac} = equivalent diameter above core = 1.34 cm .

The following condition also exists:

$$Af_c V_c = Af_{ac} V_{ac} \quad ,$$

where

Af_c, Af_{ac} = flow area through core, above core (cm^2)

V_c, V_{ac} = coolant velocity through core, above core (cm/sec) .

Thus

$$V_c = 3.375 V_{ac} ;$$

therefore

$$V_{ac} = 382 \text{ cm/sec}$$

$$V_c = 1290 \text{ cm/sec}$$

$$\Delta P_c = 5.38 \text{ atm}$$

$$\Delta P_{ac} = 0.07 \text{ atm} .$$

The temperature rise of the coolant through the core region is given by

$$\begin{aligned} \Delta T_c &= \frac{q_{Tc}}{A f_c V_c C_p \rho_w} \\ &= 18.7^\circ\text{C} . \end{aligned}$$

The heat generated in the sheath above the core is

$$q_{T_{ac}} = q_{ac}''' \rho_{al} V_{al} ,$$

where

$$q_{ac}''' = \text{volumetric heat generation above core} = 75 \text{ watt/gm} .$$

Thus

$$q_{T_{ac}} = 7.88 \times 10^4 \text{ watts} .$$

The coolant temperature rise through the sheath above the core is

$$\begin{aligned} \Delta T_{ac} &= \frac{q_{ac}}{A f_{ac} V_{ac} C_p \rho_w} \\ &= 5.8^\circ\text{C} . \end{aligned}$$

The total coolant temperature rise through the control channel is 24.44°C .

To find the maximum centerline and wall temperatures of the blade, the following equation is used:

$$t_o - t_{fc} = \frac{q_a}{2A_b k} + \frac{q_b}{A_b} \left[\frac{b}{k_c} + \frac{1}{h_c} \right] , \quad (\text{C-1})$$

where

$$\begin{aligned}
 t_{fc} &= \text{exit temperature of coolant from core section } (^{\circ}\text{C}) \\
 k, k_c &= \text{thermal conductivity of absorber and cladding} \\
 &\quad [\text{watts}/(\text{cm})(^{\circ}\text{C})] \\
 a &= \text{half-thickness of absorber (cm)} \\
 b &= \text{thickness of cladding (cm)} \\
 t_0 &= \text{temperature at center of blade } (^{\circ}\text{C}) \\
 A_b &= \text{heat transfer surface area of blade (cm}^2\text{)} \quad ,
 \end{aligned}$$

and the heat transfer coefficient is given by

$$h_c = 170(1 + 10^{-2}t - 10^{-5}t^2) \frac{V_c^{0.8}}{De_c^{0.2}} \quad ,$$

with

$$\begin{aligned}
 t &= \text{mean temperature of the coolant } (^{\circ}\text{C}) \\
 k &= 0.916 \text{ watt}/(\text{cm})(^{\circ}\text{C}) \\
 k_c &= 2.04 \text{ watt}/(\text{cm})(^{\circ}\text{C}) \\
 t &= 66.6^{\circ}\text{C} \quad .
 \end{aligned}$$

The maximum temperature at the center of the blade is found to be

$$t_0 = 105.7^{\circ}\text{C} \quad .$$

The temperature at the wall of the blade is given by

$$t_w - t_{fc} = q_b / A_c h$$

and is found to be

$$t_w = 92.2^{\circ}\text{C} \quad .$$

The temperature of the sheath wall above the core is

$$t_{w_{ac}} - t_{f_{ac}} = \frac{q_{ac}}{A_{ac} h_{ac}}$$

and

$$t_{w_{ac}} = 113.6^{\circ}\text{C} \quad .$$

The temperature at the center of the sheath above the core is found to be

$$t_{o_{ac}} = 118.7^{\circ}\text{C} \quad .$$

In Eq. (1) it is assumed that the heat generated is uniform throughout each specific region. Based on this assumption, the temperatures derived are on the conservative side and represent an extreme heating condition.

III. CONTROL BLADE ABOVE CORE

When the control blade is in its raised position, the aluminum follower is centered in the core and the poison section is in the sheath above the core. For this position, the following conditions were found to exist:

total heat generated in core section	1.89×10^5 watts
total heat generated in section above core	1.89×10^5 watts
total heat generated in blade above core	1.10×10^5 watts
total heat generated in sheath above core	0.79×10^5 watts
fluid velocity through control channel	975 cm/sec
temperature rise of coolant through core	18.7°C
temperature rise of coolant past sheath and blade above core	18.7°C
total temperature rise of coolant	37.4°C
maximum temperature at center of blade	120°C
maximum temperature at wall of blade	108.3°C

Appendix D*

SHIELDING DESIGN CRITERIA, PARAMETERS, AND CALCULATIONS

I. RADIATION DOSE DESIGN CRITERIA

The reactor plant shielding has been designed in accordance with the dose criteria listed in Table D-1. A discussion of these criteria follows.

Table D-1
RADIATION DOSE DESIGN CRITERIA

Situation	Location	Dose Rate	
		Allowable	Calculated
Reactor at power under normal conditions	(1) Inner surface of pressure vessel	$5 \times 10^{13} \text{ Mev/(cm}^2\text{)(sec)}$ $8 \times 10^9 \text{ fast neutrons (>1 Mev) (cm}^2\text{)(sec)}$	1.4×10^{13} 1.7×10^8
	(2) Inner face of magnetite concrete biological shield and also canal inner concrete surfaces	$2 \times 10^{11} \text{ Mev/(cm}^2\text{)(sec)}$	
	(3) Outer face of biological shield at shortest radius on core midplane	0.25 mr/hr	
	(4) At surface of pool water above the reactor	0.25 mr/hr	
	(5) Outer surface of primary coolant pipe tunnel for radiation entering the reactor building main floor area	0.25 mr/hr	
	(6) Control rod drive room below thick section of magnetite shield below steel shield on axial centerline		25 mr/hr 10 r/hr
	(7) Spent-fuel storage and handling canal outer surface of the walls not common to the primary coolant heat exchanger cubicle	4 mr/hr*	
	Water surface from spent cores	2 mr/hr*	1.2 mr/hr
	from primary coolant heat exchanger cubicle	2.5 mr/hr	
	(8) Heat exchanger building floor above all heat exchanger and coolant cleanup system cubicles		2.5 mr/hr
	inside surface of cubicle wall for radiation originating in adjacent cubicle		7.5 mr/hr
	walkway side of wall separating walkway from coolant cleanup systems		7.5 mr/hr
	exterior surface of primary coolant heat exchanger cubicle, building wall		7.5 mr/hr
	exterior surface of primary coolant, cleanup system cubicle building wall		2.5 mr/hr
Reactor during shutdown period under normal conditions	(1) Pool water surface above reactor from 2-hr decay core during transfer operations		15 mr/30 sec
	(2) Control rod drive room directly below steel shield 15 min after shutdown		7.5 mr/hr
Immediately after complete core meltdown where coolant is contained	(1) All locations where personnel might be present		10 r/hr
Shielding casks	(1) One meter from any readily accessible cask surface californium-252 transfer cask spent fuel shipping cask		10 mrem/hr 10 mr/hr
	(2) Cask exterior surface californium-252 transfer cask spent fuel shipping cask		200 mrem/hr 200 mr/hr

These dose rates only apply during shutdown since this is when the 2-hr decay core will be placed in the canal

*H. F. Reed, R. Gustafson, A. D. Rossin, R. H. Armstrong, J. Polonisik, R. R. Rohde, and A. E. McCarthy contributed to the work reported here.

A. Reactor Vessel

The vessel diameter is 2.44 m. Thus, there is enough water to reduce the fast neutron removal flux impinging on the vessel to 1.0×10^{17} nvt for 20 exposure years or by a factor of 50 below the design criterion of 5×10^{18} n/cm². The vessel thermal stress is small (56 kg/cm² for a 1.91-cm-thick spherical vessel) in spite of the absence of thermal shields. Gamma dose measurements at the location where the spherical vessel joins the cylindrical neck indicate that the gamma dose rate is eight times the dose rate at the same reactor radius on the core midplane. This is higher by nearly a factor of 2 than would be expected from the difference between the attenuation in the beryllium reflector and an equal thickness of water in the axial direction. Future studies must ascertain the need for thermal shielding in these vessel transition areas.

B. Biological Concrete Shield

Both magnetite and ordinary concrete should not be subjected to an impinging gamma energy flux of intensity greater than 2×10^{11} Mev/(cm²)(sec) because this overstresses these types of concrete, which are not reinforced, to the point of cracking, particularly at the impingement surface where the stress is greatest. Sufficient steel reinforcing must be utilized at critically stressed areas in the concrete to prevent the development of hairline cracks, which degrade the structural integrity.

The design dose rate established at the outer surface is $\frac{1}{10}$ of the yearly tolerance (for 40-hr/week exposure) or 0.25 mr/hr. This criterion was established for the recessed area around the horizontal through beam holes where the reactor radius is smallest, after discussion with reactor users.

C. Control Rod Drive Room

Below the reactor, a 1.70-m thickness of concrete is needed primarily for structural purposes. Magnetite concrete instead of ordinary concrete is used in this region to maintain concrete bond integrity and to minimize radiation streaming around penetrations without extensive complementary shielding.

The shield within the header is large (some 64 cm of steel). Since it is highly improbable that anyone would enter the control rod drive room except after shutdown, the dose rate criterion was established at 7.5 mr/hr at 15 min after reactor shutdown. During reactor operation, the dose rate below the steel shield (through which pass the control rod drives) is 10 r/hr.

D. Above the Reactor

It was arbitrarily decided to limit the dose received by a fuel handler during spent-fuel removal and lateral transfer operations to 15 mr. During operation, the criterion for radiation at the surface of the water shield is 0.25 mr/hr.

The walls of the primary coolant pipe tunnel extend above the main floor level of the reactor building. Therefore, the experimental area dose rate criterion applies during reactor operation, i.e., 0.25 mr/hr.

Since the transport part of the spent-fuel canal will not ordinarily be used for short-decay cores during reactor operation, no biological or experimental radiation tolerances were specified. Nevertheless, the radiation heating rate in the concrete must not exceed 8×10^{-3} watt/cc in order to prevent a thermal stress greater than the tensile strength (~ 39 kg/cm² for magnetite) of unreinforced concrete. Since this heating rate by spent fuel gammas will create a thermal stress at the inner surface of the magnetite canal wall that is just below the concrete tensile strength, reinforcing is needed.

In the storage and handling canal, only transient occupation of the adjacent areas is expected. On this basis, the dose rates at the exterior surface of the canal walls and at the surface of the canal water were specified as 4 and 2 mr/hr, respectively, for a 2-hr decay core. An additional 2.5 mr/hr is also allowed at the canal water surface for N¹⁶ radiation originating in the primary coolant heat exchanger cubicle.

E. Heat Exchanger Building

The cubicle shielding in the heat exchanger building is dependent upon induced coolant activation and coolant contamination.

During normal operation, less cubicle shielding is required if a coolant holdup tank is utilized. It is questionable, however, that the concrete savings offset the cost of the holdup tank, with its attendant space and shielding requirements. For example, increased capacities of the coolant cleanup system are needed to achieve the same water purity. Added chemical cleanup or water storage is required in the event of contamination by fission products. In areas downstream of the coolant holdup tank, where the shielding would be less than without a holdup tank, long-life contaminants from an accident would present a serious radiation problem.

Hence, the shielding is designed for no N¹⁶ decay during normal operation, so that the dose rate immediately after a core meltdown (where the coolant is contained) will not be greater than 10 r/hr in those areas which might be occupied by personnel.

The upper portion of the spent-fuel storage canal wall should be of sufficient thickness to attenuate the radiation from the heat exchanger cubicle to a value which does not exceed 2.5 mr/hr at any point on or above the canal water surface. Likewise, the dose rate criterion established for the top surface of the magnetite concrete covers is 2.5 mr/hr.

These dose rates, of course, correspond to the yearly tolerance based on a 40-hr work week. If the cubicle covers are of uniform thickness, the dose rates for points not in the region above the heat exchangers will be somewhat lower than the permissible limit. Since the floor area above the cubicle will not be a permanent work area, the dose rate criterion is considered to be conservative.

At the inside wall surface of the cleanup systems and secondary coolant cubicles, the dose rate criterion for radiation originating in an adjacent cubicle is 7.5 mr/hr. Although the combined dose rate from more than one cubicle could exceed 7.5 mr/hr, this value is used for design because only occasional access is required for maintenance.

At the floor level above these cubicles, the design level is 2.5 mr/hr.

Exposure times in the walkway beside coolant cleanup systems cubicles are of short duration and the dose rate is set at 7.5 mr/hr.

In those exterior areas where earth cannot be utilized for shielding purposes, the concrete walls will be of sufficient thickness to limit the exiting dose rate to 2.5 mr/hr at the wall surface. Because of small exposure areas, considerable earth shielding, and a complementary reactor building shielding wall, the dose rate criterion at the exterior surface of the primary coolant, heat exchanger, cubicle wall is 7.5 mr/hr.

The proposed Federal regulations regarding the shipment of radioactive materials in a single cask state, in part, that the dose at any readily accessible surface of the cask shall not exceed 200 mr/hr or 10 mr/hr at a distance of 1 meter from the cask surface. These regulations are satisfied by the design of the spent-fuel and californium-252 shielding casks.

II. CORE CHARACTERISTICS ASSUMED FOR SHIELDING CALCULATIONS

All of the shielding calculations regarding the core have been made for a solid cylindrical core having the following physical and operational characteristics:

Radius	25 cm
Height	45 cm
Specific power	2 kw/cc
Operating time	300 hr

Composition by volume per cent

Aluminum	55%
Water ($\rho = 0.97 \text{ gm/cm}^3$ at 85°C)	45%

The material properties pertinent to the shielding calculations are given in Table D-2.

Table D-2

PROPERTIES OF MATERIALS USED IN THE SHIELDING CALCULATIONS

	Water	Aluminum	Beryllium	Iron	Ordinary Concrete	Magnetite Concrete	Benelex Masonite	Paraffin	Lead
Density (g/cc)	1.0	2.7	1.84	7.6	2.34	3.52	1.4	0.9	11.4
Gamma-ray linear absorption coefficient cm^{-1}									
1	0.0706	0.166	0.103	0.47	0.150	0.212		0.0654	0.805
2.3	0.0454	0.106	0.0664	0.301	0.097	0.142		0.0420	0.498
4	0.0338	0.0837	0.0487	0.248	0.0746	0.113		0.0307	0.473
6	0.0279	0.0713	0.0387	0.227	0.0628	0.0997	0.037	0.0248	0.510
8	0.0242	0.068	0.0333	0.223	0.0569	0.0936		0.0214	0.548
0.4	0.105	0.249		0.74	0.22	0.35			2.5
0.8	0.079	0.185		0.527	0.167	0.239			1.03
1.3	0.0614	0.145		0.403	0.132	0.188			0.646
1.7	0.0532	0.125		0.348	0.114	0.163			0.558
2.2	0.0465	0.109		0.308	0.099	0.145			0.503
2.5	0.0433	0.103		0.291	0.093	0.137			0.487
2.8	0.0408	0.097		0.278	0.088	0.130			0.474
Gamma-ray energy absorption coefficient, cm^{-1}									
1	0.0314	0.0726	0.0453	0.198					
2.3	0.0254	0.0603	0.0374	0.172					
4	0.0215	0.0542	0.0308	0.169					
6	0.0192	0.0509	0.0259	0.175					
8	0.0174	0.0498	0.0233	0.185					
2.0						0.084			
Fast neutron removal cross section cm^{-1}	$\begin{cases} 0.129 \\ 0.091 \end{cases}$	0.079	0.132	0.168		0.102	0.0984		
Thermal neutron absorption cross section cm^{-1} (Maxwellian spectrum)	0.019	0.012	0.0011	0.19		0.058			
Thermal expansion $\text{cm}/(\text{cm})(^\circ\text{C})$						10.3×10^{-6}			
Modulus of elasticity kg/cm^2						0.303×10^6			
Conductivity $\text{watts}/(\text{cm}^2)(^\circ\text{C})(\text{cm})$						0.0291			
Poisson's ratio						0.15			
Capture gamma emission, Mev/capture									
1				0.2					
2.3				0.3					
4				0.96					
6				1.5					
8				3.5					
0.4	2.3	0.26				0.73			
0.8		2.8	1.5			0.96			
1.3		1.1	2.6			2.3			
1.7		2.5	2.5			2.9			

Core radiation source intensities have been calculated and are given in Tables D-3 and D-4 for operation and two hours after shutdown.

Table D-3

CORE GAMMA SOURCE INTENSITIES
DURING REACTOR OPERATION

<u>E (Mev/photon)</u>	<u>Mev/(cm³)(sec)</u>
1	6.5×10^{14}
2.3	4.5
4	1.3
6	0.29
8	0.23

Table D-4

CORE GAMMA SOURCE INTENSITIES
TWO HOURS AFTER SHUTDOWN

(Core Operation - 300 hr)

<u>E (Mev/photon)</u>	<u>Mev/(cm³)(sec)</u>
0.4	4.9×10^{12}
0.8	23.0
1.3	6.0
1.7	7.8
2.2	6.0
2.5	2.0
2.8	0.27

III. CALCULATIONS

All of the calculations for concrete shields, with the exception of the top portion of the reactor biological shield, were made for magnetite* concrete of density 3.52 g/cc.

A. Fast Neutron Removal and Thermal Neutron Fluxes

The calculation of the fast neutron removal flux in the radial direction was made by using the cylinder-to-line source equation which includes the effects of two hydrogen cross sections.¹ The source term was

*Judicious use of ordinary concrete in some sections of the shield might save money; but a comprehensive economic analysis was not possible during this study.

¹D. S. Duncan and H. O. Whittum, Jr., Application of Fast Neutron Removal Theory to the Calculation of Thermal Neutron Flux Distributions in Reactor Shields, NAA-SR-2380 (July 1, 1958).

the average number of fission neutrons produced per unit core volume. Within the core, the fast flux was determined by multiplying the first group (~ 1.35 Mev) of a four-group neutron flux core calculation by the factor 1.75 to account for the neutrons in the remainder of the fission spectrum.

The thermal neutron flux was calculated by the IBM-704 with a diffusion theory code.² This code accepts the fast neutron removal flux multiplied by the removal cross section as a point-by-point source term for the thermal neutron flux. The transfer fraction for neutrons leaving the fast group and entering the thermal group was assumed to be 1.0.

B. Core and Capture Gamma Fluxes

The core and capture gamma fluxes at various reactor radii for the preliminary shielding arrangement are listed in Table D-5.

The gamma flux at the core center was calculated by means of Rockwell's equation, including buildup for the flux on the axial centerline of a solid cylindrical source. Effective axial power distribution factors up to 1.2 for the lowest energy group were applied and are included in the source intensities listed in Tables D-3 and D-4. Radial power distribution factors and the presence of the center flux trap are neglected. The core gamma fluxes at the surface of the core were set equal to half the values at the core center.

The core gamma flux within the shields was found by utilizing Rockwell's cylinder-to-line-source conversion equation. The two-term exponential buildup factor for water was incorporated in the equation. Generally, for materials more dense than water, this type of buildup is too high for low-energy gammas and too low for high-energy gammas. It is believed that the error in buildup for high-energy gammas does not exceed the predicted value by a factor of 2 throughout the shields. Since an opposite error occurs when linear buildup is used for the capture gamma calculations, the effects of using the water buildup factors are somewhat nullified.

The capture gammas originating in as many as three sections of each shield region were calculated separately by machine, with the assumption of the existence of a single exponential thermal neutron source.^{3,4} Since the capture gamma equations are based upon a slab source geometry, it was necessary to multiply the boundary source term of each section or region by a slab-to-annulus-source geometrical correction factor.

²M. K. Butler and J. M. Cook, RE-34, An IBM-704 Reactor Shielding Program, ANL-5859 (June 1959).

³Therese Czaja, Private Communication.

⁴M. Grotenhuis, Lecture Notes on Reactor Shielding, ANL-6000 (March 1959).

Table D-5

REACTOR MIDPLANE GAMMA FLUXES ON 5-Mw/LITER CORE OPERATION (PRELIMINARY SHIELD ARRANGEMENT)

[N B A number x xxE ± yy to read (x xx)10^{±yy}]Gamma Flux, Mev/(cm²)(sec)

Reactor Radius (cm)	Gamma Energy Group (Mev/photon)	Gamma Source Region with Outer Radius in cm					Total at Radius
		Core 25	Reflector 56	Water Zone 122	Steel-Water Zone 156	Magnetite 426	
0	1	1 26E16					1 26E16
	2 3	0 83					0 83
	4	0 24					0 24
	6	0 054					0 054
	8	0 043					0 043
		2 43E16					2 43E16
(Calculations based on solid core)							
25	1	6 2E15					6 2E15
	2 3	4 1	1 75E14				4 28
	4	1 2	0 75				1 28
	6	0 27	1 48				0 42
	8	0 22	1 53				0 37
		1 2E16	5 51E14				1 26E16
40	1	6 32E14					6 32E14
	2 3	6 85	1 29E14	5 31E12			8 19
	4	2 27	0 546				2 82
	6	0 55	1 06				1 61
	8	0 467	1 09				1 56
		1 65E15	3 99E14	5 31E12			2 05E15
56	1	1 18E14					1 18E14
	2 3	1 65	3 54E13	4 82E13			2 49
	4	0 693	1 73				0 866
	6	0 182	3 68				0 55
	8	0 155	3 98				0 553
		3 86E14	1 29E14	4 82E13			5 64E14
80	1	1 46E13					1 46E13
	2 3	3 52	0 626E13	9 31E12	0 856E7		5 08
	4	1 65	0 403		2 06		2 05
	6	0 484	0 99		5 09		1 47
	8	0 451	1 17		17 8		1 62
		7 57E13	3 19E13	9 31E12	2 58E8		1 17E14
122	1	0 602E12					0 602E12
	2 3	3 21	0 573E12	7 27E11	0 718E8		4 51
	4	2 18	0 585		1 05		2 77
	6	0 853	1 81		1 97		2 66
	8	0 799	2 48		5 79		3 28
		7 64E12	5 45E12	7 27E11	9 53E8		1 38E13
139	1	0 39E10					0 039E11
	2 3	11 3	0 161E11	1 06E10	0 15E8	0 528E4	1 4
	4	14 7	0 294		0 251	1 5	1 76
	6	6 85	1 16		0 507	5 06	1 85
	8	7 23	1 68		1 52	6 95	2 4
		4 05E11	3 3E11	1 06E10	2 43E8	1 4E5	7 45E11
156	1	0 0245E9					0 0245E9
	2 3	3 92	0 477E9	3 12E8	0 0963E7	0 258E6	4 71
	4	9 73	1 62		0 205	0 396	11 4
	6	5 48	8 17		0 467	1 04	13 7
	8	6 41	12 5		1 45	1 33	18 9
		2 56E10	2 28E10	3 12E8	2 22E7	3 02E6	4 87E10
246	2 3	0 11E5		5 44E2		0 743E2	0 116E5
	4	2 63	0 373E5		0 386E2	1 92	3 01
	6	4 31	6 2		2 95	7 47	10 5
	8	8 17	16 2		16 0	12 1	24 4
		1 57E6	2 28E6		1 93E3	2 22E3	3 8E6
336	4	0 99E1	0 104E1		0 105E-2	0 213E 1	0 11E2
	6	4 83	5 66		2 65	1 39	1 05
	8	13 4	25 4		24 8	3 21	3 89
		1 97E2	3 12E2		2 76E 1	4 81E-1	5 05E2
426	4	0 612E 3	0 033E-3				0 645E-3
	6	7 6	5 52				13 1
	8	32 7	44 4		4 23E-5	7 35E-5	77 1
		4 09E-2	5 0E-2				9 08E-2

The correction factor was found by dividing the section or region source boundary radius by the radius of the point where the capture gamma flux was desired. It is understood that the source of the entire region or section is changed in a like manner so that the exponential slope of the source remains the same for capture gamma flux calculations at all reactor radii.

The core gammas and the reflector capture gammas comprise essentially all of the gamma flux. Beyond the water region, a change in either the shield thickness or the shield material, insofar as it is not a change to a thick gamma shield, will not significantly reduce the dominance of core and reflector gammas. It was therefore easily possible to extrapolate the results of the preliminary shield design (see Table D-5) which was more than adequate to satisfy the criterion for radiation leakage.

All that was necessary was to establish what change would be desirable with respect to shielding mediums and their arrangement. It appeared that a region of water placed between the vessel and the concrete offered the most benefits. Answers were sought as to the necessity of a concrete thermal shield region of steel and water, its thickness, and the thicknesses of the water and concrete regions.

The method involved solution of a set of three simultaneous equations, one of which stated that the total available shielding thickness was equal to the sum of its parts. The remaining two were exponential attenuation equations related to the design radiation criteria at the inner and outer surfaces of the concrete. Exponential slopes of flux curves were obtained from the preliminary calculations for the steel-water and concrete regions. Since water was to be the added shielding material, the exponential slope for this region was set equal to the gamma linear absorption coefficient for 7-Mev gammas in water. In spite of spectrum hardening in the water, this method gives conservative results because the average gamma energy exiting from the 34-cm steel-water region of the preliminary arrangement was 5.1 Mev/photon. This average energy is even less for smaller thicknesses in the steel-water region. After the first iteration, the gamma flux slopes were changed in agreement with the calculated region thicknesses and a second calculation was made. Once the shield thicknesses were determined, the method was checked against the cylinder-to-line-source conversion equation and found to be satisfactory.

C. Beam Hole Shielding Calculations

In the beam hole, the fast neutron removal flux was calculated by the Simon-Clifford equation and multiplied by a factor of 10 as suggested by Rockwell and also by a single exponential factor for attenuation through the shields. This method was also used for calculating the thermal neutron flux at the first shield surface.

The calculation for gamma rays was made by using Rockwell's equation for a cylindrical duct and a Fermi emitter coupled with the exponential attenuation factor through the shields. It was discovered, however, that in order to obtain agreement with experimental gamma flux measurements, the flux from this equation, like the Simon-Clifford equation, also had to be multiplied by a factor of 10.

D. Primary Coolant Heat Exchanger Cubicle Shielding Calculations

The N^{16} gamma flux at the top surface of the magnetite shield above the second in-line heat exchanger was estimated, by means of Rockwell's upper limit equation, for the flux exterior to the end of a cylindrical source. The main source is the upper heat exchanger header which is assumed to have a diameter of 1.32 m and a height of 91.4 cm. The N^{16} coolant activity at this location during normal operation is 1.1×10^6 dis/(ml)(sec), having undergone an 18-sec decay since leaving the core. A 1.07-m magnetite shield is necessary to satisfy the dose rate criterion of 2.5 mr/hr at the floor level. The dose from the adjacent heat exchangers was found by converting them to line sources 2.13 m long. This is the height of the adjacent heat exchanger which emits radiation reaching the point where the dose rate was calculated without being shadow shielded by the second heat exchanger.

One of the places where personnel are likely to be exposed to the highest radiation in the event of a complete core meltdown is the secondary system pump and pipe cubicle. In order to estimate the dose rate in this location, a comparison is made with the fuel element meltdown at the Westinghouse Test Reactor.⁵

Insofar as fission product decay power is concerned, the effective WTR core power before the incident was about 35 Mw, as compared with a power level of 155 Mw for the AARR core operating at 2 Mw/liter. The effective number of fuel subassemblies in the core was 74.4, including the nine control rod fuel followers at their operating positions.

The entire subassembly was affected by the meltdown. The active lengths at the top of the elements, where the cladding was unharmed, varied from 7.6 cm for the center element to 20 cm for the outermost element. Sectioning showed that some of the meat in these areas had become molten and run out. The remainder of the subassembly was missing or showed evidence of having been in the molten state. The melted subassembly is assumed to have a maximum-to-core average power generation factor

⁵WTR Staff, Report on WTR Fuel Element Failure, April 3, 1960, WTR-49 (July 7, 1960).

of 1.5, which includes the radial maximum-to-average power factor of from 1.33 to 1.45 plus the effect of a greater than average fuel concentration. The fission products were distributed in a water coolant having a volume of 454,000 liters, which is 8 times the coolant volume in the AARR design.

The intensity of radiation emanating from the 76.2-cm-diameter downcomer pipe below the WTR head tank resulted in a dose rate of 40 r/hr on contact 3 hr after the meltdown occurred. The WTR report indicated the primary coolant activity was decaying at a rate of $t^{-1.2}$. Apparently factors such as coolant flow time between the core and the head tank, gaseous separation from the coolant, fission product mixing with the coolant, and quantity, quality, and rate of fission product leakage influenced the radiation level of the head tank because, even for infinite operation, the total fission product power decays by more than a factor of 3.7 during a shutdown period of 3 hr. Several of these variables cannot be distinguished accurately, so they are not considered to be significantly different for the WTR and AARR systems insofar as coolant activity immediately after a meltdown is concerned.

The melted fuel subassembly had a continuous and total reactor operating history of about 30 hr at power before the incident occurred, but a meltdown in the AARR could occur at an irradiation up to 10 times longer. For coolant activity at short decay times, the longer irradiation is more severe only when it is not possible for the bulk of the short half-lived radioactive isotopes (those which are at equilibrium for both the short and long irradiation) to enter the coolant stream. Time did not permit the investigation of fission product escape mechanisms; let it simply be assumed that the maximum coolant activity is independent of irradiation time. This hypothesis is essentially valid when the irradiation time is more than a day, the decay time less than 10 min, and the coolant activity is representative of the core activity as a result of coolant-suspended core particulate matter.

With the above information, it is possible to make a rough calculation of the maximum dose rate at the surface of a 76-cm pipe after a complete AARR core meltdown. The dose rate is $3.7 \times 50 \times 8 \times 155 / (35 \times 1.5) / 74.4 = 2.6 \times 10^5$ r/hr.

A calculation was performed to determine the dose rate at the 76-cm pipe surface immediately after shutdown for a complete AARR core meltdown in which the fission product inventory is assumed to be equally dispersed throughout the coolant. The gamma flux, including buildup, was calculated at the center of the pipe for a source strength of 6.07×10^{11} Mev/(ml)(sec) at an energy of 1.0 Mev/photon. The surface flux was assumed to be $\frac{1}{3}$ of the flux at the center of the pipe. The resulting dose was 1.17×10^7 r/hr or 45 times the value calculated by comparison to the WTR incident. It is obvious that the 1.17×10^7 -r/hr dose rate would be even

greater if some of the fission products were assumed adhering to the pipe wall. An opposite and possibly greater effect is that of decay time needed for the fission products to disperse and reach maximum specific activity within the coolant.

E. Radiation from Cf²⁵² Production Sample

The conditions for the production of californium-252 in the AARR is given in Table D-6.

Table D-6

CONDITIONS FOR THE PRODUCTION OF CALIFORNIUM-252 IN THE AARR

Specific power	1.26 Mw/liter
Initial Sample	100 g Cm ²⁴⁴
Perturbed thermal neutron flux	
in sample	$2.5 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$
Thermal neutron flux depression	40%
Total perturbed pile neutron flux	$4.1 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$
Exposure time	2 yr

The sample is a plate-type assembly similar to the fuel box subassembly. The dimensions of the sample were assumed to be 6.0 cm² by 30.5 cm in height. In order to allow sufficient space for the entire sample, the cask chamber dimensions were set at: diameter - 13.0 cm; height - 34.3 cm.

To establish cooling requirements, it is necessary to know the energy release rate from the sample, both during irradiation and when it is placed in the shielding cask.

Material balance equations were written for the two isotopes (Cm²⁴⁴ and Cm²⁴⁵) which release the greatest amount of specific energy during the 2-yr irradiation. The capture gamma energy release was assumed to be 8 Mev/capture, whereas the fission energy releases of Cm²⁴⁵ and Cf²⁵² were set at 105 and 110 per cent, respectively, of the energy release in U²³⁵ fission. These factors tend to account for the greater kinetic energy of fission fragments for these isotopes.^{6,7} The results of the calculations are presented in Table D-7.

⁶W. H. Sullivan, Trilinear Chart of Nuclides, U. S. Gov't. Printing Office, Washington 25, D. C. (Jan 1947).

⁷Glenn T. Seaborg, The Transuranium Elements, New Haven, Yale University Press (1958).

Table D-7

ENERGY RELEASE AND ISOTOPIC CONCENTRATION FOR
Cf²⁵² PRODUCTION SAMPLE

Isotope	Time*	Concentration x 10 ²¹ (atoms/sample)	Energy Source (watts/sample)			
			Capture Gammas	Alpha Decay	Neutron or Spontaneous Fission	Fission Product Decay
Cm ²⁴⁴	A	230	18000	260		
	B	4.7		5.1		
	C			5.1		
Cm ²⁴⁵	A	3.2	2100		430000	
	B	0.064				610
	C					61
Cf ²⁵²	A					
	B	0.17		1.6	2.0	
	C			1.6	2.0	

*A - 11.1 days (Cm²⁴⁵ maximum concentration)

B - 2.0 yr (shutdown after assumed continuous operation)

C - 6 hr after shutdown

It is seen that there is not sufficient heat emission after a shutdown period of 6 hr to warrant the installation of a system to cool the cask. Since the decay of Cm²⁴⁵ fission products is responsible for most of the heat emission, a greater heat liberation occurs at shorter irradiation times. The peak concentration of Cm²⁴⁵ atoms is greater than the concentration at the end of 2-yr irradiation by a factor of 50. Thus, if it is assumed that the Cm²⁴⁵ fission product decay power is also increased by this factor, a considerably longer decay time (perhaps several weeks) would be necessary for a sample removed around the time of Cm²⁴⁵ peak concentration. If it is important to remove the sample and place it in the cask after short irradiation times, the problem of heat emission and heat removal should be more carefully determined.

With a knowledge of the cask type required, an investigation of shielding materials was undertaken with respect to economy, reliability, and shielding effectiveness.

In view of the radiation emitted by the sample, the shielding would have to be partially hydrogenous for neutron attenuation and, in addition, dense enough to attenuate the accompanying gamma radiation. Such a criterion is most economically satisfied by the application of concrete; however, in this case, almost 1.1 m of magnetite would be needed for the wall of the cask. Thus, with magnetite both the size and the weight of the case would be undesirable.

It was thought that the shield thickness would be minimized if a good fast neutron scatterer and gamma-ray absorber, such as steel, were combined with a good neutron moderator, such as paraffin. Because large-scale laminated shields are too expensive to fabricate, steel in the form of pellets was investigated.

In the case of a neutron moderator the two most important factors to consider regarding shield reliability are decomposition and containment. A medium such as water in conjunction with steel punchings would not safely satisfy either of these. Water would corrode the steel. It has not yet been possible to utilize corrosion inhibitors under stagnant conditions because the inhibitors tend to settle out. The other hazard associated with water is the ever-present possibility of development of a leak which could go undetected. Oil is another good moderator, one which would not attack the steel, but it too might be difficult to contain.

Paraffin appears to be the best choice since it can be obtained in high densities (0.93 g/cc) and high melting points (68°C). Because of this high melting point, the paraffin will be protected against liquefaction when the cask is placed in the hot sunlight. Paraffin does have a high volumetric coefficient of expansion, which explains the slump upon solidification. A space for expansion of the paraffin will be provided and a vacuum-pressure-relief valve connected.

Petroleum products can usually be exposed to radiation dosages in the range of 10^6 to 10^8 rads without appreciable effect on properties which will affect their end use.⁸ It is not known how the shielding properties of paraffin will be affected by irradiation, but the useful life is assumed to end with an exposure of 10^7 rads. Since the highest dose rate received by the paraffin is about 5×10^3 rads/hr, the useful shielding life of the paraffin toward the cask center is 83 days of continuous exposure. Such a longevity will suffice because of the short duration of exposure for on-site sample transportation. For longer durations, it may be desirable to increase the thickness of the cask liner and hence reduce the radiation intensity to which the paraffin is exposed.

With a 4% greater fast neutron removal cross section, polyethylene is used as the neutron moderator in the cask shielding plugs instead of high-density masonite.

By using polyethylene with a high density of from 0.94 to 0.97 g/cc, the heat-distortion temperature, which is a measure of the

⁸E. F. Baxter, Jr., and F. J. Black, The Application of Petroleum and Petroleum Products as Shielding in Nuclear Propelled Tankers, ESSO-MA- 2 (Aug. 1, 1959).

dimensional stability under stress, is 60°C to 82°C at 4.6 kg/cm².⁹ The body weights of the disks in the lower plug will result in a load of about 0.4 kg/cm² on the bottom polyethylene disk. Thus, there should be little tendency for the polyethylene to creep upwards around the edges of the disks.

To calculate the size of the steel-punching-paraffin shield, Carlson's SNG transport theory code (written for the IBM-704) was used. An 18-group cross-section set which emphasizes the low energy end of the spectrum was utilized. With 3.82 neutrons liberated per Cf²⁵² spontaneous fission, the source totalled 2.1×10^{11} neutrons/(sample-sec).^{7(p. 153)}

To allow for the radiation channeling around the steel punchings, a calculation was made of the effective fast neutron removal cross section for uniformly sized steel cylinders (height = diameter = 2 cm) having a volume fraction of 0.6.¹⁰ With the remainder of the shield (40%) assumed to be void, the effective cross section or steel volume fraction was found to be reduced by 8%, and this value was used in the SNG calculations. Subsequently, this type of calculation was found to be incorrect in that the paraffin cross section was disregarded. However, if one considers the reduced shield effectiveness created by the perturbation of punching distribution at the cask boundaries, the net error in the neutron dose rate resulting from a too small neutron cross section is not significant and is pessimistic.

For gamma-ray attenuation, radiation channeling and reduced thickness due to edge perturbations were found to be more dependent upon the size of the steel punchings. This is explained by the fact that steel is more opaque to gammas than is paraffin. Consequently, the upper limit established for both the height and the diameter of the steel punchings is 2.54 cm. The paraffin, which occupies 0.4 of the shield, was assumed to have the chemical formula CH₂.

The input for this shielding problem involved specifying a k calculation for a cylinder (S₄) with a Q option giving the source for each neutron group at each mesh point.

The source was specified as 1 fission neutron/(cm³)(sec) at all mesh points in the sample and was distributed over the neutron energy group structure according to the U²³⁵ fission neutron energy spectrum. In all of the other regions, the source was zero.

⁹Plastic Properties Chart, Modern Plastics Encyclopedia, Issue for 1961, Vol. 38, No. 1A, Breskin Publications, Inc., Bristol, Conn. (Sept 1960).

¹⁰W. R. Burrus, Radiation Transmission Through Boral and Similar Heterogeneous Materials Consisting of Randomly Distributed Absorbing Chunks, ORNL-2528 (Jan 18, 1960) p. 11.

The material specifications for the problem are given in Table 8.

Table D-8

MATERIALS AND REGIONS OF THE Cf²⁵² CASK
FOR THE SNG CALCULATION

Region	Constituent	Volume Per Cent	Outer Radius (cm)	Mesh Spacing (cm)
1 (sample)	Al	55	3.6	0.9
2	Fe punchings	60*	83.6	1.0 (first 30 cm)
	CH ₂	40		2.0 (last 50 cm)
3	Fe punchings	60*	89.6	0.4
	Boroffin**	40		
4	Fe	100	98.6	1.0

*Effective volume assumed to be 55%.

**20 wt % B₂O₃ plus 80 wt % paraffin.

The plug shielding in the axial direction was devised by comparing fast neutron removal flux calculations in both the radial and axial directions. The gamma flux calculations were done in much the same fashion as were the reactor gamma flux calculations. All calculations were done for an air-filled cask chamber. Water in this chamber reduces the neutron and gamma dose rate at the cask surface by a factor of 2.

IV. GAMMA DOSE RATE MEASUREMENTS

Gamma dose rates were determined in the second AARR critical assembly by means of photographic dosimetry. This assembly was heavily loaded with U²³⁵ and poisoned with stainless steel strips in order that it might be representative of (with respect to reactivity effects and neutron fluxes) the AARR core at the beginning of the core cycle. At the time the experiment was conducted, the critical facility had the composition depicted in Table D-9.

The photographic measuring device consisted of a DuPont film packet (Type 544)¹¹ enveloped in waterproof tape and completely surrounded

¹¹E. I. DuPont de Nemours & Co., Photo Products Dept., Wilmington 98, Delaware, Dosimeter Film, Booklet No. A-1000-a.

by 0.476-cm Lucite plates (one on each side, but none along the edges), 0.127-mm cadmium sheets, and 0.318-cm boral plates in that order. The dimensions of the constructed pack are:

Length	6.7 cm
Width	5.7 cm
Thickness	2.1 cm .

The function of the Lucite, cadmium, and boral was to achieve electronic equilibrium, gamma-ray-energy-independent film response, and the minimization of capture gamma and beta production (primarily within the emulsion, but also in the neutron attenuator), respectively.¹²

Table D-9

COMPOSITION OF AARR CRITICAL FACILITY
C-2 PRIME

Height of fuel zone, cm	50.8
Volume of fuel zone, liters	100.9
Volume of internal thermal column, liters	8.2
Material densities in fuel zone, g/cc	
U^{235}	0.0719
Stainless Steel (12 strips/cluster)	0.240
H_2O	0.384
Aluminum	1.495

Control films within the above-described packs were exposed to a standard radium gamma source having a dose rate of 1 r/hr at a radius of 20 cm. These films were developed simultaneously with the experimental films. The control film densities were then determined on an extended range Ansco-Macbeth densitometer in order to obtain a calibration curve of net film density versus film exposure in roentgens. The exposure dose of the experimental films is derived from the calibration curve and converted to a dose rate by consideration of the core power density, operating time, and shutdown time before film removal. The spatial dose rates are listed in Table D-10 in the order of increasing distance from the core center. The location of the center of the film is defined by utilizing an (x,y,z) coordinate system with the origin as the core center. The X-axis depicts the E-W direction (east positive), the Y-axis the N-S direction (north positive), and the Z-axis the vertical (up positive). All results have been obtained with the film pack described above unless noted otherwise. The particular region where each exposure was taken is also provided. The S-W quadrant of the facility used for these measurements is shown in Fig. D-1.

¹²M. Ehrlich and S. H. Fitch, Photographic X- and Gamma-ray Dosimetry, Nucleonics, 9 (3) 5-17 (1951).

Table D-10

GAMMA DOSE RATES IN AARR CRITICAL FACILITY (C-2 PRIME)

Location (x,y,z), cm	Region	Distance from Core Center, cm	Dose Rate, ^(a) roentgens/hr per milliwatts/l	Non-Standard Film Covering
0,0,0	Flux Trap	0	9 0-A, 7 7-B	
4 1, 5 9, 0		7 2	8 3-A	
-5 9, -4 1, 0		7 2	8 0-A	
±3 0, -5 9, 3 5		7 5	6 3-B, 6 8-B	
±3 0, -5 9, -3 5		7 5	9 2-B, 7 9-B	Cadmium, Cd-boral
-6 9, 4 1, 0		8 0	12 8-A, 11 5-B	Bare
-4 1, 6 9, 0		8 0	9 9-A, 10 6-B	"
14 0, -3 5, ±3.3	Fuel Zone	14 8	13 8-A, 13 8-A	"
21 0, -3 5, ±3 3		21 5	13 2-A, 12 4-A	"
14 0, 17 5, ±3 3		22 6	12 5-A, 11 7-A	"
0, 0, 25 4	Flux Trap	25 4	2 84-B	
-29 6, 0, -0 8	Beam Tube(b)	29 6	4 43-A	
9 9, 23 9, 17 8	Reflector	31 4	4 10-A	
14 6, -3 5, 29 4	Water	33 0	4 47-B	Lucite
4 1, -14 0, 29 9		33 3	2 90-A	
14 6, 3 5, 29 9		33 5	3 07-B	
29 0, ±2 9, 17 8	Reflector	34 2	3 47-A, 3 54-A	
9 9, -14 0, 29 9	Water	34 5	3 09-A	
20 4, -3 5, 29 4		36 0	4 09-B	Lucite
20 4, 3 5, 29 9		36 4	2 82-B	
±2 9, 28 5, 28 4		40 3	3 54-A, 3 47-A	Lucite
-29 0, ±2 9, 28 4		40 7	1 84-A, 2 02-A	
-42 3, 0, -0 8	Beam Tube(c)	42 3	0 84-C	
±2 9, -58 8, 0	Water	58 9	0 6-C, 0 59-C	Bare
59 8, ±2 9, 0		59 9	0 317-C, 0 291-C	
0, -58 8, 21 7		62 7	0 470-C	Bare
48 6, -43 5, 0		65 2	0 435-C	Bare
±3 9, -82 6, 0		82 7	0 080-C, 0 077-C	
±2 9, -101 8, 0		101 8	0 0302-C, 0 0302-C	
70 1, -73 9, 0		101 8	0 0275-C	
73 9, -70 1, 0		101 8	0 0265-C	
-102 9, 0, -0 8	Beam Tube(d)	102 9	0 069-C	
-66 7, +2 9, 101 6	Water(e)	121 6	0 094-C, 0 084-C	
-, -, 50 8	(f)	51 0	0 498-C	
-, -, 76 2		76 3	0 251-C	
-, -, 101 6		101 7	0 145-C	

(a) The letters A, B, and C designate the experimental run

(b) Measurement taken in the west beam tube with only the end of the tube between the film pack and the core face

(c) Same as (b) except preceded by 12 7 cm of beryllium

(d) Same as (b) except preceded by 15 2 cm of beryllium and one film pack

(e) This point is the same distance from the core center as the spherical vessel to cylindrical neck transition region

(f) The film packs for these three measurements were placed horizontally such that their centers were 4 1 cm from the core axial centerline and each pack offered minimum shadow shielding to the packs above

FIG. D-1
SOUTHWEST QUADRANT OF AHFR CRITICAL FACILITY

Additional information pertaining to the experimental runs is given in Table D-11.

Table D-11

CORE 2 OPERATING CHARACTERISTICS FOR GAMMA DOSIMETRY EXPERIMENTS

Run	Thermal Neutron Flux at Point 32 on Core Midplane, $n/(\text{cm}^2)(\text{sec})$	Calculated Core Power Level, watts	Core Operating Time, hr	Core Shutdown Time Prior to Film Removal, hr
A	4.866×10^6	0.774	0.667	0.25
B	5.490×10^6	0.873	0.667	0.25
C	2.146×10^8	34.14	0.267	1.50

In computing the absolute thermal neutron flux, a cadmium ratio of 1.67 was used and also a spectrum energy correction of 0.5783 was applied to the gold activation cross section of 96 b. The dose rate is based upon the core power density, which was found by dividing the above core powers in milliwatts by the core volume, 100.9 liters.

In order that the dose rates will typify the core after extended operation, they have been multiplied by a factor of 1.11 for runs A and B and 1.087 for run C. This factor is defined as $(0.4R + 0.6)^{-1}$, where R is the ratio of the equilibrium fission product power integrated over the core

operating time during the experimental run.^{13,14} The fraction 0.4 is the approximate contribution of fission product gammas to the total gammas resulting from fission.

For ease of comparison to the calculations, the hand-calculated gamma dose rate (derived from Table D-3) at the horizontal midplane of a solid cylindrical core (radius - 25 cm, height - 45 cm) is plotted in Fig. D-2 along with some of the experimental data from Table D-2.

FIG. D-2
MEASURED AND CALCULATED DOSE RATES FOR AARR

¹³J. R. Stehn and E. F. Clancy, Fission Product Radioactivity and Heat Generation, Proc. of the 2nd UN Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 3, 726-746 (1958).

¹⁴J. F. Perkins and R. W. King, Energy Release from the Decay of Fission Products, Nuclear Sci. and Eng., 4, 726-746 (June, 1958).

For the curve designated "along core midplane" all measurements were within 3.3 cm of the midplane. The portion of this curve within the fuel zone was obtained by dividing the bare film values of the dose rate in this zone by 1.3. The factor 1.3 represents the ration of the bare to covered (standard) film dose rates in the flux trap region and includes a small estimate (last digit dropped) of the effects on this ratio attributed to the fuel zone ratio of the thermal neutron to gamma-ray fluxes which is smaller than in the flux trap region.

The dashed curve is the estimated dose rate in a core having the above-described dimensions and a central flux trap of 7.7-cm equivalent radius such that the fuel zone volume is 80 liters. It is seen that the maximum of this curve is about 90% of the calculated value at the center of the core without a flux trap. It will also be observed that the maximum is 1.76 times greater than the maximum of the measured dose rate curve.

The calculated dose rate is for a core which is smaller and contains no stainless steel to depress the thermal flux (and hence the capture gamma source) or reduce the photon mean free path and hence, the total gamma flux, as compared with these effects in the larger core operating at an equal power density. Calculation of the total gamma flux at the center of these two cores indicated the smaller core should have a dose rate which is greater by a factor of 1.36 than the more opaque core containing the stainless steel. Applying the factors 1.36 and 1.06 to the maximum measured dose rate in the 100.9-liter core brings this value to within 82% of the maximum estimated value for the smaller core. This difference would likely be even less were it not for the higher temperature (relative to the monitor films) in which the control films were exposed and also to latent image fading of these films, since they were not developed as soon after exposure as were the monitor films.

The dose rates have been divided by the core power density rather than the total core power, since within and close to the core the gamma dose is somewhat independent of core size. However, at large distances from the core, the effect of core size becomes noticeable, as can be observed in Fig. D-2, in which the measured dose rate exceeds the calculated dose rate. A portion of this effect is also attributed to the larger percentage of high-energy capture gammas arising in the stainless steel core as compared with the smaller core. These high-energy capture gammas are more penetrating in a shield of low atomic number.

V. GAMMA AND FAST NEUTRON HEAT GENERATION

The gamma and fast neutron radiation heating rates are listed for several materials as a function of reactor radius in Table D-12.

Table D-12

HEAT GENERATION RATES FOR SMALL SAMPLES ALONG
THE AARR HORIZONTAL MIDPLANE AT A CORE POWER
DENSITY OF 2 Mw/LITER

Region	Reactor Radius, cm	Material	Heat Generation, watts/gm		
			Gamma	Fast Neutron	Total
Flux trap	0	H ₂ O	72.8	53.2	126
		Al	63.8	12.5	76
		Fe	64.0	9.1	73
		Cd	65.0 ^(a)	6.0	71
		Pb	96.0	4.3	100
Fuel zone	14	H ₂ O	97.0	88.8	186
		Al	85.1	20.8	106
		Fe	85.3	15.2	101
		Cd	86.7	9.9	97
		Pb	128	7.2	135
Fuel zone- reflector boundary	25	H ₂ O	48.6	50.4	99
		Al	42.6	11.8	54
		Fe	42.7	8.6	51
		Cd	43.4	5.6	49
		Pb	64.0	4.1	68
Beryllium reflector	40	H ₂ O	8.40	0.84	9.2
		Al	7.52	0.20	7.7
		Fe	7.82	0.14	8.0
		Cd	8.31	0.09	8.4
		Pb	11.6	0.07	11.7
Reflector- water boundary	56	H ₂ O	2.22	0.06	2.28
		Al	2.01	0.01	2.02
		Fe	2.13	0.01	2.14
		Cd	2.29	0.01	2.30
		Pb	3.11		3.11

(a) The cadmium capture gammas are not included.

The values of the gamma heating have been derived from the gamma fluxes presented in Table D-12. In the fuel zone the heating rate was assumed proportional to the estimated gamma dose rate of Fig. D-2, i.e., the maximum heating rate was taken to be 90% of that calculated at the center of a solid (without a flux trap) core while the heating rates at the core center and the core edge were set at 75% and 50%, respectively, of the maximum heating rate.

The fast neutron heating was determined by multiplying the fast neutron removal flux by the fast neutron removal cross section and assuming an energy release of 2 Mev for each neutron removed.

As a first approximation, the radiation heating rate along the axial centerline of the core is assumed proportional to the axial gamma dose rate curve of Fig. D-2.

Calculations were made of the heat flux at the surface of a control plate in the region of greatest radiation heating, i.e., on the core midplane about 14 cm from the core center. The heat flux was found to be 80 watts/cm² at each surface of the 1.5-mm cadmium plate having a 0.5-mm aluminum cladding. The percentages of the heat flux attributed to fast neutron removal collisions and cadmium capture gammas is about 12 and 3%, respectively. A likely position for the control plates is at the outer radius of the core. If the plate centers are 1.0 cm from the fuel zone, the midplane heat flux would be only 30 to 40% of the above value.

In addition to the above comparisons, an energy balance check can be made. The core gamma-ray sources used are listed in Table D-13.

Table D-13

CORE GAMMA RAY SOURCES

Source	Energy, Mev/fission
Prompt fission	7.83
Equilibrium fission products	7.22
U ²³⁵ capture gammas	1.18
Aluminum capture gammas	0.916
Water capture gammas	0.535
TOTAL	17.681

As can be seen from the table, the total gamma energy production in the core is 17.7 Mev/fission, which, when multiplied by the factor 1.17 (the factor which correlates the average energy midplane gamma flux based on uniform power distribution to the effects of the axial maximum to average power distribution factor of 1.28), becomes 205 watts/cm³ maximum calculated. The heat absorption at the core midplane is $[0.55 \times 2.7 \times 85.1 \text{ (aluminum)}] + [0.45 \times 0.97 \times 97 \text{ (water)}] + [0.04 \times 144 \text{ (U}^{235}\text{)}] = 126 + 42 + 6 = 174 \text{ watts/cm}^3$. If a solid core is considered, this becomes $(1.11 \times 174 \times 100)/205 = 94\%$ of the gamma energy produced.

The energy associated with fast neutron production at the core midplane is about 5.0 Mev/fission or 63.5 watts/cm³, including the 1.28 axial

power distribution factor. The fast neutron energy absorption is $[0.55 \times 2.7 \times 20.8] + [0.45 \times 0.97 \times 88.8] + [0.04 \times 6.4] = 30.9 + 38.8 + 0.3 = 70 \text{ watts/cm}^3$ which, when corrected for a solid core, is about 20% greater than the neutron energy produced. It is concluded from these comparisons that the total heat generation rates listed in Table D-13 are accurate to within $\pm 15\%$.

ACKNOWLEDGMENTS

The AARR critical experiments were conducted by members of the Experimental Physics Section of the Reactor Engineering Division, in particular, by Q. L. Baird, J. W. de Villiers, J. O. Juliano, C. N. Kelber, R. Kiyose, and K. E. Plumlee.

N. Peters helped with fuel element flow tests and associated instrument development work. E. Groh was responsible for the construction of test sections and the perfection of plate fastening techniques.

Contributions to the reactor design were made by G. Popper, R. Stover, and A. B. Heineman.

L. E. Link gave advice and data accumulated from the previous high-flux reactor studies.

Finally, credit is due E. Barts and C. N. Kelber for a careful review of the report draft and their acceptance of many editorial chores.